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ORNL/TM-13551

Strategy for the Future Use and Disposition of Uranium-233: History, Inventories, Storage Facilities, and Potential Future Uses

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LOCKHEED MARTIN ENERGY RESEARCH CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL-27 (3-96)

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ORNL/TM-13551

STRATEGY FOR THE FUTURE USE AND DISPOSITION OF URANIUM-233: HISTORY, INVENTORIES, STORAGE FACILITIES, AND POTENTIAL FUTURE USES

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June 1998

* Managed by Lockheed Martin Energy Research Corp., under contract DE-AC05-96OR22464 for the U.S. Department of Energy.

PREFACE

This report is one in a series of reports which examines issues associated with the future use and disposition of ^{233}U. A brief description of the other reports is included herein.

ORNL/TM-13550, *Strategy for the Future Use and Disposition of Uranium-233: Overview*. This document is a summary of the path forward for disposition of surplus ^{233}U. It includes required activities, identifies what major programmatic decisions will be required, and describes the potential disposition options.

ORNL/TM-13551, *Strategy for the Future Use and Disposition of Uranium-233: History, Inventories, Storage Facilities, and Potential Future Uses*. This document includes the sources, historical uses, the current inventory of ^{233}U, and potential future uses. The inventory includes quantities, storage forms, and packaging of the material.

ORNL/TM-13552, *Strategy for the Future Use and Disposition of Uranium-233: Technical Information.* This document summarizes scientific information on ^{233}U. This includes production methods, decay processes, and material characteristics. The requirements for storage and disposal are also included.

ORNL/TM-13553, *Strategy for the Future Use and Disposition of Uranium-233: Options*. This document describes the proposed disposition alternatives, identifies what material in inventory could be treated by each disposition option, and provides an initial analysis of each option. A listing of the legislative or regulatory changes required for each alternative is also provided.

ORNL/TM-13524, *Isotopic Dilution Requirements for ^{233}U Criticality Safety in Processing and Disposal.* This document determines and defines how much depleted uranium must be mixed with ^{233}U to prevent the potential for nuclear criticality under all expected process and disposal facility conditions.

ORNL/TM-13517, *Definition of Weapons-usable Uranium-233.* This document develops a definition of non-weapons-usable ^{233}U to provide a technical basis for changing the safeguards and security requirements for storing, using, and disposing of ^{233}U that is isotopically diluted with depleted uranium (DU).

ORNL/TM-13591, *Uranium-233 Waste Definition: Disposal Options, Safeguards, Criticality Control, and Arms Control.* This document develops a definition of what ^{233}U-containing materials are waste and what ^{233}U-containing materials are concentrated fissile materials.

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ACRONYMS AND ABBREVIATIONS

AEA	Atomic Energy Act of 1954
AEC	United States Atomic Energy Commission
ANL-E	Argonne National Laboratory–East, Argonne, Illinois
ANL-W	Argonne National Laboratory–West, Idaho Falls, Idaho
ASB	Air Support Building
BAPL	Bettis Atomic Power Laboratory
BIO	Basis for Interim Operations
BMU	breeder mock-up
CEUSP	Consolidated Edison Uranium Solidification Program
CH	contact-handled (transuranic waste)
COG	Cell Off-Gas
COMED3	Commonwealth Edison Company, Dresden Reactor Unit 3, Morris, Illinois
CY	calendar year
DAW	dry active waste
DNFSB	Defense Nuclear Facilities Safety Board
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DU	depleted uranium
D&D	decontamination and decommissioning
EPA	U.S. Environmental Protection Agency
ERDA	U.S. Research and Development Administration (succeeded AEC; now DOE)
ES&H	Environmental Safety & Health
FAB	Facility Authorization Basis
FHU	fuel handling unit
FRC	fuel receipt criteria
FSAR	Final Safety Analysis Report
FSC	fuel storage container
FSV	Fort St. Vrain
FSVR	Fort St. Vrain Reactor, Platteville, Colorado
FY	fiscal year
GA	General Atomics
GBOG	Glove Box Off-Gas
GTCC	Greater-than-Class-C (low-level waste)
HEU	highly enriched uranium
HLW	high-level waste
HTGR	high-temperature, gas-cooled reactor
IAEA	International Atomic Energy Agency
ICPP	Idaho Chemical Processing Plant
ID	inner diameter
IDMS	isotope dilution mass spectrometry
ILTSF	Intermediate Level Transuranic Storage Facility
IN	identification number

INEEL	Idaho National Engineering and Environmental Laboratory
KAPL	Knolls Atomic Power Laboratory
LANL	Los Alamos National Laboratory
LBNL	Lawrence Berkeley National Laboratory
LEU	low-enriched uranium
LLNL	Lawrence Livermore National Laboratory
LLW	low-level waste
LWBR	light-water breeder reactor
LWR	light-water reactor
MLLW	mixed low-level waste
MOX	mixed oxide
MSRE	Molten Salt Reactor Experiment
NAC	Nuclear Assurance Corporation
NBL	New Brunswick Laboratory
NEPA	National Environmental Policy Act of 1969
NFS	Nuclear Fuel Services
NMMSS	Nuclear Materials Management and Safeguards System
NRC	U.S. Nuclear Regulatory Commission
NRF	Naval Reactors Facility (INEEL), Idaho Falls, Idaho
NSR	national security requirements
OD	outer diameter
OR	Oak Ridge (Reservation)
ORNL	Oak Ridge National Laboratory
PF	power-flattening
PNNL	Pacific Northwest National Laboratory
ppm	parts per million
PPU	planned programmatic use
PVC	polyvinylchloride
R&D	research and development
RCRA	Resource Conservation and Recovery Act
RDF	Radiochemical Development Facility
RFETS	Rocky Flats Environmental Test Site
RH	remote-handled (transuranic waste)
RWMC	Radioactive Waste Management Complex
SNF	spent nuclear fuel
SNM	special nuclear material
SRE	Sodium Reactor Experiment
SRS	Savannah River Site
SS	stainless steel
TRU	transuranic
TSA	Transuranic Storage Area
USQD	Unreviewed Safety Question Determination
VOG	Vessel Off-Gas
WAC	waste acceptance criteria
WIPP	Waste Isolation Pilot Plant, Carlsbad, New Mexico
ZPR	Zero-Power Reactor

CHEMICAL ELEMENTS

Ac	actinium
Am	americium
Bi	bismuth
F	fluorine
Na	sodium
Np	neptunium
O	oxygen
Pb	lead
Po	polonium
Pu	plutonium
Rn	radon
Th	thorium
Tl	thallium
U	uranium

EXECUTIVE SUMMARY

This document provides background information on the man-made radioisotope ^{233}U. It is one of a series of four reports that map out potential national strategies for the future use and disposition of ^{233}U pending action under the National Environmental Policy Act (NEPA). The scope of this report is separated ^{233}U, where separated refers to nonwaste ^{233}U or ^{233}U that has been separated from fission products. Information on other ^{233}U, such as that in spent nuclear fuel (SNF), is included only to recognize that it may be separated at a later date and then fall under the scope of this report. The background information in this document includes the historical production and current inventory of ^{233}U, the uses of ^{233}U, and a discussion of the available facilities for storing ^{233}U.

During the 1960s, over 2 t of ^{233}U was produced in the United States for possible use in nuclear weapons and for generating electricity. Most of this material is currently in storage at a handful of sites across the U.S. Department of Energy (DOE) complex (Table ES.1). About 99% of the ^{233}U inventory resides at two sites: Oak Ridge National Laboratory (ORNL) and Idaho National Engineering and Environmental Laboratory (INEEL). ORNL is the national repository for ^{233}U and already has about half of the inventory in a variety of chemical and physical forms. Upgrades to aging ventilation and confinement systems at ORNL are underway or are planned in order to assure safe storage. INEEL also has a large fraction of the inventory as unirradiated fuel pellets, rods, and assemblies. Improvements to storage configurations are underway or are planned in order to continue INEEL's role in the storage of unirradiated fuel forms. These planned and ongoing improvements are being integrated into the complex-wide, systematic evaluation of safe ^{233}U storage in response to Recommendation 97-1 from the Defense Nuclear Facilities Safety Board (DNFSB).

Because cost and risk can typically be reduced by consolidation, it may be advantageous to consolidate the small percentage of material scattered throughout the DOE complex at one or two of the aforementioned sites. Other sites would require significant upgrades to accommodate ORNL or INEEL holdings because of the unique storage characteristics of ^{233}U with regard to ventilation, shielding, and remote handling requirements.

In addition to consolidating the inventory, isotopic dilution and repackaging of surplus materials may also be desirable. Isotopic dilution with depleted uranium (DU) can prevent ^{233}U use as a nuclear weapon. Dilution would also reduce safeguards and security costs in storage and transportation. During the dilution process, material could be converted into more stable forms and repackaged. Currently, there is no standard package for ^{233}U, although one is being developed. Several different types of packaging might be used depending on the intended use or final disposition.

DOE Defense Programs (DP) has declared much of the stored material as excess to national security needs. However, there are several potential uses for ^{233}U. The use with the greatest potential is using ^{213}Bi, a decay product of ^{233}U in radioimmunotherapy. Because ^{233}U is fissile, it can also be used as fuel in nuclear reactors or as a weapons device. One current small-scale use for ^{233}U is as a calibration spike in safeguards procedures for nuclear materials.

The considerations for what fraction of the current inventory should be preserved for future use depend on several issues. First, ^{233}U always contains a small amount of the contaminant isotope ^{232}U. The decay products of ^{232}U are highly radioactive and require special handling. The current inventory has a variety of qualities with regard to ^{232}U content, ranging from 1 to about 200 ppm (on a total uranium basis). It is preferable to use ^{233}U with the minimum amount of ^{232}U in all applications. The second issue pertains to other isotopes of uranium mixed in with the ^{233}U, specifically ^{235}U and ^{238}U. A large portion of the inventory has a high quantity of ^{235}U associated with it. The presence of bulk amounts of ^{235}U complicates storage because of the added volume needing safeguards and criticality controls. Isotopic dilution using DU may remove safeguards and criticality concerns, but it increases the overall mass and may limit applications that depend on the fissile nature of ^{233}U. The third issue concerns the packaging of the material. There is no standard packaging (although one is being developed); consequently, the inventory exists in a variety of packages. For some applications, the material will have to undergo processing which may be complicated (or facilitated) by the variety of packaging forms.

Table ES.1. ^{233}U materials in the United States

Site	U-233 Material Grouping (in kg ^{233}U)[a]			
	Unirradiated		Wastes[b]	SNF[c]
	Pure	Commingled[d]		
Argonne National Laboratory-East	*			
Argonne National Laboratory-West	0.2			
Bettis Atomic Power Laboratory (BAPL)	0.4			
Brookhaven National Laboratory	*			
Dresden Reactor SNF				0.3
Fort St. Vrain Reactor SNF[e]				236.0
General Atomics (GA)	*			
Hanford Site	0.1		8.3	
INEEL		34.2[f]	59.0	
Fort St. Vrain Reactor SNF				90.1
Light-Water Breeder Reactor (LWBR, Shippingport)				
A. Irradiated LWBR fuel (SNF)				523.7
B. Unirradiated LWBR fuel		317.4[g]		
Peach Bottom Reactor SNF				46.3
Knolls Atomic Power Laboratory	*			
Lawrence Berkeley National Laboratory	*			
Lawrence Livermore National Laboratory (LLNL)	3.3			
Los Alamos National Laboratory (LANL)	7.1		4.4	
Mound Advanced Technology Center	*			
Nevada Test Site			0.2	
New Brunswick Laboratory	*			
ORNL	326.0	101.2[h]	13.3	
Molten Salt Reactor Experiment (MSRE) SNF				31.0[i]
Miscellaneous SNF				147.5
Pacific Northwest National Laboratory (PNNL)	*			
Rocky Flats Environmental Technology Site (RFETS)	*			
Savannah River Site (SRS)				
Dresden Reactor SNF				15.4
Elk River Reactor SNF				14.7
Sodium Reactor Experiment				1.1
Irradiated Thorium Target Slugs				*[j]
Y-12 Plant		0.8[k]		
TOTALS (subject to round-off differences)	337.1	453.6	85.2	1106.1

a An * represents a quantity between 0.5 and 50 g

b Based on DOE 1997 (for contact-handled and remote-handled transuranic wastes only)

c Based on DOE 1994

d Commingled unirradiated ^{233}U materials are mixed with other radionuclides (e.g., natural Th, ^{235}U), which comprise the majority of heavy-metal content.

e Material stored in Colorado managed by INEEL SNF program.

f Material declared as waste and located at INEEL's Radioactive Waste Management Complex

g LWBR fuel mixed with 14 t of natural thorium as oxides

h Includes Consolidated Edison Uranium Solidification Program CEUSP oxide material (101.1 kg ^{233}U) mixed with 796 kg ^{235}U and 145 kg of other uranium isotopes plus 0.1 kg ^{233}U mixed with 2 metric tonnes of natural Th as a nitrate solution

i SNF from the MSRE is currently being recovered and added to the ORNL inventory

j Material mixed with 55 kg natural thorium

k Material mixed with 39 kg ^{235}U

1. HISTORY OF THE PRODUCTION OF ^{233}U

1.1 HISTORY

During World War II, the United States obtained exclusive control of about 90% of the world's known (at that time) supply of high-grade uranium and thorium ore used in the making of atomic bombs. One of the chief sources for ore was the Shinkolobwe mine in the Belgium Congo. It was believed that controlling the supply of ore would limit the ability of other countries to build atomic bombs (Rhodes 1995). Indeed, the known world supply of uranium ore at that time was fairly scarce as compared to that of thorium.

During 1944, it was recognized that it might be possible to use another isotope of uranium, ^{233}U, to make atomic bombs. Uranium-233 is even rarer than ^{235}U, but it can be produced readily by transmutation of relatively abundant thorium. Thorium is obtained by the refinement of monazite sand, of which there were major deposits in Brazil and, more importantly, in North Carolina and South Carolina. Uranium-233 can be bred from thorium in a nuclear reactor much as plutonium is bred from ^{238}U, and, like plutonium, ^{233}U can then be chemically separated from thorium much more easily than ^{235}U can be physically separated from ^{238}U (Rhodes 1995). While the discovery of ^{233}U was too late for it to be considered as a nuclear weapons material in World War II, starting in the 1950s, it was investigated for this application.

In the 1960s, ^{233}U was investigated as a nuclear reactor fuel. The thorium fuel cycle has important advantages compared to the uranium/plutonium fuel cycle. First, thorium is more abundant than uranium. Second, the thorium fuel cycle produces far fewer long-lived radioisotopes, such as ^{237}Np. Also, very little plutonium is produced and thus the risk of weapons proliferation is lessened. However, a disadvantage is that thorium does not readily undergo fission, so the thorium fuel cycle is not self-sustaining. Unlike ^{238}U, which breeds plutonium, thorium is not naturally present in economical reactor fuels. Finally, some of the thorium is transformed into ^{232}U, which has a decay product, ^{208}Tl, which emits a highly energetic gamma ray when it decays. This complicates the handling of Th-^{233}U based fuel.

1.2 U. S. PRODUCTION

During the 1950s, the Atomic Energy Commission (AEC) directed that 30 kg of ^{233}U be produced at Hanford. During this time, Oak Ridge National Laboratory (ORNL) began development of a Th-^{233}U extraction process that later became known as the Thorex process. The use of ^{233}U in nuclear weapons was investigated. As it turned out, however, ^{233}U bombs were complex because of the presence of and intense radiation associated with another rare uranium isotope, ^{232}U (Rhodes 1995).

By the 1960s, the U.S. interest in ^{233}U had expanded to include the possibilities that ^{233}U would be useful in nuclear reactors used to produce electricity. Between 1964 and 1970, the AEC directed the production and recovery of ~2000 kg of ^{233}U at the Hanford site and the Savannah River Site (SRS). Hanford ^{233}U production was conducted in two distinct campaigns: the first in 1966 (Atlantic Richfield Hanford Company 1968) and the second in 1970 (Jackson and Walser 1977). The SRS produced ^{233}U during five different campaigns between 1964 and 1969 (Severynse 1996). Table 1.1 provides a summary of ^{233}U production between 1964 and 1970. This table does not include production of ^{233}U that may have resulted from ^{233}U-Th fuels in commercial reactors that were not reprocessed to recover the uranium.

Additional ^{233}U was produced in several commercial reactors: Indian Point I, Elk River, Dresden, Peach Bottom, and Fort St. Vrain (FSV). The spent nuclear fuel (SNF) from most of these reactors was not reprocessed and remains in storage at U.S. Department of Energy (DOE) facilities at the Idaho National Engineering and Environmental Laboratory (INEEL) and the SRS. Of the commercial reactors, the Indian Point Reactor is noteworthy in that this SNF was reprocessed to recover its ^{233}U and ^{235}U components.

Table 1.1. Production of separated ^{233}U between 1964 and 1970

Campaign	^{233}U (kg)	References
Hanford 1966	234.85[a]	Atlantic Richfield Hanford Company 1968
Hanford 1970	628[b]	Jackson and Walser 1977
Subtotal for Hanford	863	
SRS—233U-1; 3/64–5/64	64	Orth 1979
SRS—233U-2; 12/64–2/65	76	Orth 1979
SRS—Thorex-1; 10/65–1/66	169	Orth 1979
SRS—Thorex-2A; 4/68–7/68	78	Orth 1979
SRS—Thorex-2B; 7/69–11/69	177	Orth 1979
Subtotal SRS	564[c]	
Indian Point Reactor	126[d]	Martin Marietta Corporation 1984
Totals	1553	

a Of the 234.85 kg of ^{233}U produced, ~223 kg were recovered, and ~12 kg were sent to the Hanford tank farm (Tank 102-C).

b Of the 628 kg of ^{233}U produced, ~597.5 kg were recovered, and ~33 kg were sent to the Hanford tank farm (Tank 104-C).

c No information was available to determine how much, if any, of the reported SRS production was discharged to waste tanks.

d It is estimated that 105 kg were recovered.

During the early 1960s, the AEC sponsored a test irradiation of the ThO_2-UO_2 fuel core in the Consolidated Edison Indian Point-1 Reactor (ORNL Chemical Technology Division 1981). The Indian Point fuel was made with highly enriched uranium (HEU) and thorium and, thus, contained very little ^{238}U and ^{239}Pu. This fuel was reprocessed at the Nuclear Fuel Services (NFS) plant at West Valley as Lot 11 in 1968 (U.S. DOE 1996a). In 1968 and 1969, this material was transferred to ORNL as uranyl nitrate solution and placed into the Thorium Reactor Uranium Storage Tank in Building 3019 at ORNL. In the late 1970s, a project called the Consolidated Edison Uranium Solidification Program (CEUSP) was initiated to solidify the material for safe long-term storage. Following the design and installation of processing equipment in the early 1980s, solidification began in April 1985 and was completed in April 1986. Since that time, the solidified CEUSP material, containing ~1043 kg of U [76.5 wt % ^{235}U, 9.67 wt % ^{233}U, 1.4 wt % ^{234}U, 5.63 wt % ^{236}U, 6.84 wt % ^{238}U, and 120 parts per million (ppm) ^{232}U], has been in storage at Building 3019 at ORNL (ORNL Chemical Technology Division 1987). This single batch of material is (a) about half of the material volume of uranium that contains ^{233}U in the U.S. inventory and (b) the only batch with significant quantities of other separated uranium isotopes comprising the majority of the material.

In the same time period, the Light-Water Breeder Reactor (LWBR) Program was initiated to demonstrate the potential use of ^{233}U in commercial breeder reactors. A full reactor core of ^{233}U fuel was fabricated from $^{233}UO_2$ prepared at ORNL and natural thorium oxide prepared at the National Lead plant in Fernald, Ohio. Fuel blending, pelletizing, and rod and assembly fabrication were performed at the Bettis Atomic Power Laboratory (BAPL) in West Mifflin, Pennsylvania. High quality ^{233}U (low ^{232}U content) was used to fabricate LWBR fuel assemblies to minimize radiation exposures to workers. The ^{233}U-Th core was irradiated in the Shippingport Reactor and used a large fraction of the total ^{233}U separated inventory in the United States. This material was not reprocessed for recovery of the ^{233}U. An additional partial reactor core also was prepared to allow refueling of the Shippingport Reactor. However, the program ended before this fuel was used. The fresh fuel is stored along with the SNF from the core at INEEL. This is the largest batch in the inventory in terms of mass of ^{233}U.

1.3 WORLDWIDE PRODUCTION

In the middle of the 1950s, a perceived shortfall in long-range availability of uranium as a nuclear fuel focused more attention on the thorium fuel cycle. Thorium extraction plants were built in many countries. Germany and France each separated about 2000 t of thorium, part of which is still available (Lung and Gremn 1996). It is also believed that the Soviet Union produced ^{233}U in amounts comparable to that of the United States during the cold war.

However, by the middle of the 1980s, the success of light-water reactors (LWRs) and an increase in the world uranium supply caused a waning interest in thorium fuel cycles. Today, only India, which has the world's largest thorium reserves, continues to pursue a thorium fuel cycle. Recently, the Kamini reactor, a 30-kW research reactor in Kalpakkam, attained criticality (*The Hindu* 1996). Kamini is the only currently working reactor that uses separated ^{233}U as fuel. Research on a small scale continues in a number of countries [France, Germany, South Africa, China (Jiahua and Borong 1994), Russia, and Japan].

2. CURRENT INVENTORY OF ^{233}U

2.1 SEPARATED MATERIAL

An accountable quantity of ^{233}U materials has been identified at 17 domestic sites. The inventory of separated ^{233}U in the United States totals about 790 kg and is contained in about 1500 packages. (Here, *separated* ^{233}U refers to nonwaste ^{233}U or ^{233}U that has been separated from fission products, and *packages* refers to external containers.) A summary of current ^{233}U material characteristics and inventories are presented in Table 2.1 (for major inventory sites) and in Table 2.2 (for small-quantity sites). These tables exclude irradiated ^{233}U, which falls under the SNF program, and waste that is not considered part of the accountable inventory (see Table ES.1). A detailed summary of current ^{233}U material characteristics and inventories for major inventory sites is reported in Appendix A.

Table 2.1. Summary, by large-quantity site, of domestic ^{233}U material characteristics and inventories[a]

Site	No. of packages	Total U (kg)	^{233}U (kg)	^{235}U (kg)
Argonne National Laboratory-West (ANL-W)	63	0.155[b]	0.154[b]	0
BAPL	13	0.427	0.405	0.014
INEEL	213	358.6	351.6	0
Lawrence Livermore National Laboratory (LLNL)	50	3.321	3.253	0
Los Alamos National Laboratory (LANL)	109	7.243	7.105	0
ORNL	1054	1387.7	427.3	796.3
Oak Ridge Y-12 Plant	5	42.6	0.8	38.7
Totals (subject to roundoff differences)	1507	1800.0	790.6	835.0

a Excludes materials that are categorized as waste or SNF that may be candidate ^{233}U materials after recovery or separation.

b Includes cumulative contribution from 62 packaged items having less than 0.1 kg.

The inventory falls naturally into several batches for reasons of quality, composition, or packaging. These batches will be discussed in the following sections. In cases for which the ^{232}U content at the time of separation is known, a decay calculation has been made to give the current quantity of ^{232}U. This should not be taken to mean that the hazard from ^{232}U has decreased. Instead, the opposite is true because of the ingrowth of decay products of ^{232}U such as ^{208}Tl. However, before future use, the material could undergo further processing to separate ^{233}U and ^{232}U from their decay products.

Table 2.2. Summary, by small-quantity site, of domestic ^{233}U material inventories and characteristics[a]

	Site[b]	Total U (kg)	^{233}U (kg)	^{235}U (kg)
Argonne National Laboratory-East (ANL-E)	5	0.028[c]	0.028[c]	0
Brookhaven National Laboratory (BNL)	*d*	*d*	0.002	*d*
General Atomics Laboratory (GA)	4	0.172	0.031	< 0.001
Hanford Site	3	0.597	0.079	0
Knolls Atomic Power Laboratory (KAPL)	26	*d*	<0.010	*d*
Lawrence Berkeley National Laboratory (LBNL)	*d*	*d*	0.031	*d*
Mound Plant	1	*d*	<0.005	*d*
New Brunswick Laboratory (NBL)	3	*d*	0.005	*d*
Pacific Northwest National Laboratory (PNNL)	15	0.048	0.047	0
Rocky Flats Environmental Test Site (RFETS)	5	0.008	0.008	0
Totals	>30	>0.853	<0.246	>0

a Based on Ives. May 11, 1998 and June 8, 1988. Excludes materials that are categorized as waste or SNF that may be candidate ^{233}U materials after recovery or separation.
b Sites listed are those containing less than 0.1 kg ^{233}U.
c Excludes 0.008 kg of reactor irradiated material.
d Information not reported by site.

2.1.1 CEUSP Material

Uranium from the Consolidated Edison Indian Point Reactor fuel was recovered by the NFS plant at West Valley, New York, and sent to ORNL as a liquid for storage. During 1985 and 1986, this material was thermally converted from nitrate solution to a U_3O_8 monolith in storage cans. The CEUSP material contains 101 kg of ^{233}U and 796 kg of ^{235}U in a total of 1043 kg of uranium.

The uranium in the CEUSP material was separated from fission products during 1968 through 1969. At the time of processing at ORNL it had an isotopic composition of 140 ppm ^{232}U (123 ppm in 1998), 9.7 wt % ^{233}U, 1.4 wt % ^{234}U, 76.5 wt % ^{235}U, 5.6 wt % ^{236}U, and 6.8 wt % ^{238}U. This is one of only two groups of material that contains an appreciable amount of fissile material other than ^{233}U. The neutron poisons cadmium and gadolinium were added to the CEUSP material to reduce the risk of a criticality accident during its 17-year period of storage as a liquid. The presence of these two elements could complicate disposition planning for the CEUSP material since it may be classified as a mixed waste under the Resource Conservation and Recovery Act (RCRA). However, such a classification is unlikely since the materials in question were added as part of the process.

The CEUSP material is stored in over 400 stainless steel (SS) inner cans that are welded shut (Fig. 2.1). The inner cans have 3.4-in. outer diam. and 24.25-in. lengths. Each inner can is placed inside a double-seamed, tin-plate outer canister. The outer canisters have 3.625-in. interior diam. and lengths of

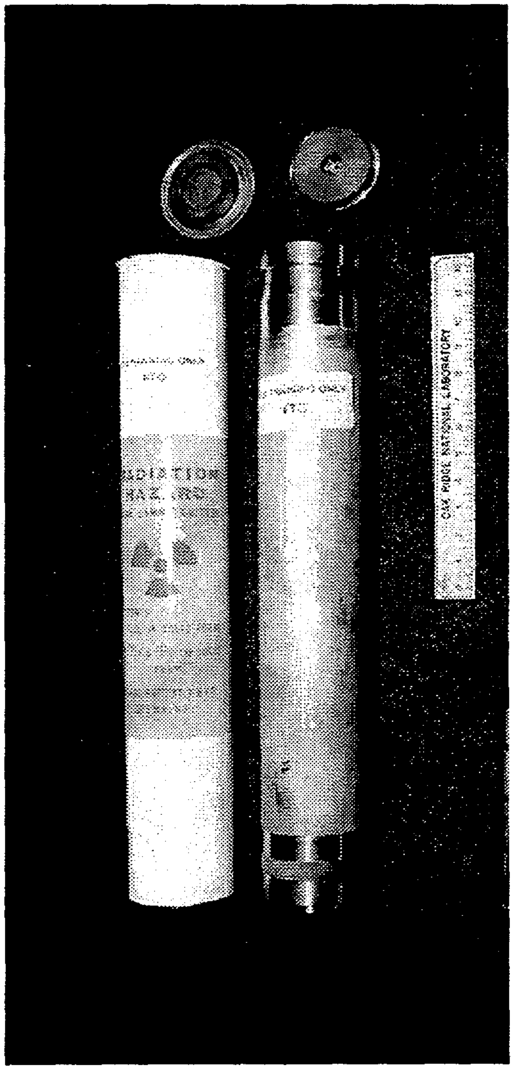

Fig. 2.1. CEUSP containers.

24.75 in. (Martin Marietta Corporation 1984). The CEUSP material was packaged by placing each container in a high-temperature furnace and then adding ^{233}U as a nitrate solution. The package was baked at a temperature of ~800°C for 3 h (McGinnis, et al. 1987). In the package, the nitrate decomposed to an oxide, forming a cast-in-place monolith. While this approach makes for a very stable storage form, disposition is complicated by the fact that this material is integral with its package.

2.1.2 Low-Quality Powder

ORNL is storing UO_3 powder from the SRS. This material was separated during 1964 and 1965 and consists of 67.4 kg of total uranium with a composition of 220 ppm ^{232}U (158 ppm in 1998), 91 wt % ^{233}U, 7.3 wt % ^{234}U, 1 wt % ^{235}U, 0.08 wt % ^{236}U, and 0.3 wt % ^{238}U. This material is considered low grade because of its high ^{232}U content. There are no other bulk contaminants in this material.

The low-quality ^{233}U is stored in 140 packages. The inner cans are made from aluminum and are welded shut. The outer cans are also fabricated from aluminum and have welded lids. Certain disposition options may take advantage of this packaging by dissolving the can along with the material.

2.1.3 Radioactive Waste Management Complex (RWMC) Rods and Pellets

The RWMC at INEEL is storing unirradiated fuel pellets and rods containing ^{233}U. There are 35.1 kg of uranium containing 34.2 kg ^{233}U. The rod and pellet material is stored in 2R containers inside 172 55-gal or 110-gal U.S. Department of Transportation (DOT)-6M shipping drums.

The ^{233}U at RWMC had 5 to 10 ppm ^{232}U in 1970. This corresponds to a range of 3 to 8 ppm in 1998. There are two batches of material at RWMC which also contain thorium. The first batch is composed of rods and loose pellets in bottles. The rods contain 5.42 kg total uranium, 5.27 kg ^{233}U, and 502 kg thorium. The pellets contain 426 g total uranium, 415 g ^{233}U, and 39 kg thorium. The second batch contains 196 fuel rods, two short rods, and pellets in slip-top cans. These rods contain 1.66 kg total uranium, 1.63 kg ^{233}U, and 53 kg of thorium.

The RWMC material is in a variety of forms that might require repackaging for economical long-term storage. There are fuel rods in steel tubing and zirconium-clad fuel rods. The loose pellets are in SS tubes with an o-ring seal. All the material is contained in 2R containers inside 55- and 110-gal DOT-6M drums.

2.1.4 ORNL Monolith

Material from SRS, 60.3 kg of ^{233}U in a total of 65.2 kg of uranium, was run through the CEUSP process after it was delivered to ORNL. This material was separated before April 1966 and was converted to a U_3O_8 monolith in 1986.

Unlike the CEUSP material, the ^{235}U content of this material is only 0.90 wt %. The ^{232}U composition at the time of processing was only 20 ppm (18 ppm in 1998). However, cadmium was added to this material to simulate CEUSP feed conditions, thereby facilitating smooth process operations.

This material is stored in 27 welded, SS cans placed in tin-plated, double-seamed overpack canisters. Like the CEUSP material, disposition options may be complicated by the integral nature of this material with its container and the presence of the RCRA-regulated material, cadmium.

2.1.5 UO_x Powder in Screw-Top Cans

A total of 96.6 kg of uranium is stored at ORNL as UO_2 or U_3O_8 powder with 91.3 kg of ^{233}U. This high-quality material was separated at the SRS and Hanford site during the 1960s and processed to its final form at ORNL during the period 1980 to 1985. This material had ^{232}U contents between 5 and 9 ppm at the time of processing at ORNL. This range corresponds to a range of 4 to 8 ppm in 1998.

The oxide lots are stored in 174 screw-top, SS cans that are 3 3/8-in. in diam and either 3-in. or 7-in. tall. The primary cans are packed into 7-in. or 8-in. tall, double-seamed, tin-plated cans, with one or two primary cans per isotope can. The top half of Fig. 2.2 shows the primary cans, while the secondary containers are shown in the bottom half.

2.1.6 Zero-Power Reactor (ZPR) Material

A total of 45 kg of high-quality ^{233}U in 46.2 kg of total uranium as U_3O_8 prepared for the ZPR experiment is being stored at ORNL. This uranium was separated in the 1960s at the SRS and Hanford site and processed to its final form at ORNL from 1978 to 1979.

At the time of the processing at ORNL, the ZPR material had a ^{232}U content ranging from 6 to 9 ppm. Accounting for decay puts this concentration range between 5 and 8 ppm in 1998. The ZPR material also has small quantities of some other uranium isotopes (1.2 wt % ^{234}U, 0.08 wt % ^{235}U, 0.018 wt % ^{236}U, and 0.68 wt % ^{238}U).

The ZPR material is contained in a total of 1743 packets, which are hollow, rectangular SS shells that are nickel-plated and welded shut. Each packet is 3 in. long, 2 in. wide, and 0.25 in. thick containing approximately 33 g of U_3O_8 powder. The packets are placed inside 128 double-seamed, tin-plate cans, with diameters of 3.875 in. and heights of 8 in.

Fig. 2.2. Typical ^{233}U storage containers.

2.1.7 LWBR Fresh Fuel

A partial spare core and fuel elements prepared by BAPL between 1972 and 1977 for the Shippingport LWBR are being stored at the Idaho Chemical Processing Plant (ICPP) at INEEL. There are 40 containers of seed and blanket rods containing 306.64 kg of total uranium, of which 300.80 kg are ^{233}U.

Additionally, there is an assembled seed module with 16.84 kg total uranium, of which 16.56 kg are ^{233}U. All this uranium was separated originally at the SRS and Hanford Site and processed to a ceramic-grade $^{233}UO_2$ at ORNL during the period 1972 to 1976. Further description of the unirradiated LWBR fuel is provided in Appendix B.

The fresh ^{233}U fuel at ICPP had a composition of 5 to 10 ppm ^{232}U at the time of fuel fabrication. This corresponds to a range of 4 to 8 ppm in 1998. There are no RCRA-regulated materials in any of this fresh fuel inventory, but all of the batches contain extensive amounts of thorium oxide (ThO_2). The intact rods contain 16.1 t of thorium, while the spare seed module contains 0.43 t of thorium. This ^{233}U at ICPP is well-characterized, welded-clad, pelletized material adequate for long-term storage.

2.1.8 Y-12 Material

The Y-12 Plant in Oak Ridge is storing 42.6 kg of uranium oxide and metal with 0.8 kg of ^{233}U and 38.70 kg of ^{235}U. This material was separated around 1967.

The ^{232}U content of the Y-12 material was between 5 and 10 ppm at the time of separation. This corresponds to a range of 4 to 7 ppm in 1997. This material is unusual because of its high ^{235}U concentration. The relative amount of ^{233}U to ^{235}U is very close to the ratio where the inhalation or ingestion doses of ^{233}U (with ^{232}U) and ^{235}U (with ^{234}U) are equal. The material at Y-12 is stored in DOT-6M 110-gal drums.

2.2 UNSEPARATED MATERIALS

2.2.1 Inventory

In addition to the separated material discussed previously, there is considerable ^{233}U in SNF and in waste from ^{233}U processing. This material is not considered part of the ^{233}U inventory, but may be recovered and included in the future. Table 2.3 is a listing of the inventory of the ^{233}U in SNF.

2.2.2 Possible Reasons for Separation

There are several reasons why unseparated ^{233}U may be recovered. In the case of the only material currently being recovered, the Molten Salt Reactor Experiment (MSRE) at ORNL, there is an unresolved safety issue. In other cases, there may be criticality issues associated with placing ^{233}U wastes in a

Table 2.3. Inventory of ^{233}U in spent nuclear fuel from nuclear power generation and development[a]

	Location[b]	Composition[c]	Description	Estimated burnup (MWd/MTIHM[c])	U content, kg		
					Total	^{235}U	^{233}U
Dresden Unit 1 Reactor SNF	COMED		18 irradiated fuel rods		0.5	0.2	0.3
	SRS	UO_2–ThO_2, SS-clad	Intact assemblies in 4.4- × 4.4- × 135-in. cans	4,000–10,000	684.00	37.54	15.39
Elk River Reactor SNF	SRS	UO_2–ThO_2, SS-clad	Assemblies, 3.5 × 3.5 × 81.62 in.	50,000 (max.)	224.29	186.16	14.72
Fort St. Vrain Reactor SNF	FSVR	U–Th carbide and Th carbide, pyrolytic carbon-coated particles in graphite matrix	1464 fuel elements,[d] 14 × 16 × 31 in.		822.4	404.5	236.0
	INEEL	U–Th carbide and Th carbide, pyrolytic carbon-coated particles in graphite matrix	744 fuel elements,[d] 14 × 16 × 31 in.	6,000–26,000	308.33	167.65	90.14
Light-Water Breeder Reactor (Shippingport) Irradiated fuel (SNF)	INEEL	UO_2 ceramic fuel pellets with Th, Zr, and Ca oxides; Zr-clad; Th blanket	48 elements, 10 by 103 in. in 24 SS cans,25.5 by 158 in.	Unknown	656.64	10.56	523.68
Molten Salt Reactor Experiment (MSRE)	ORNL	LiF–BeF_2–ZrF_4–UF_4	*e*	Unknown	36.95	0.940	31.01
Peach Bottom Unit 1 SNF	INEEL	U–Th carbide, pyrolytic carbon-coated particles in graphite matrix	1,603 graphite elements, 3.5 by 144 in. in 90 Al cans 4.5 by 153 in.	30,000 Core I 60,000 Core II	332.42	233.54	46.31
Sodium Reactor Experiment (SRE)	SRS	U, Th rods, SS-clad	cans, 3.5-in.-diam by 110.25-in. length	10,000	154.93	143.41	1.05
Miscellaneous SNF[f]	ORNL	U metal chunks, UO_x powder, and UO_2	Stored in SS cans	Varies	159.16	0	147.48
Total					3,379.6	1,184.5	1,106.08

[a]Based on U.S. DOE 1994.
[b]Current storage location of ^{233}U material.
[c]Metric tons initial heavy metal
[c]Fuel composition and cladding material, where applicable. Zr = Zircaloy.
[d]A fuel assembly for a high-temperature, gas-cooled reactor (HTGR), viz., fuel rods inserted into hexagonal graphite blocks.
[e]The MSRE project was concluded in 1969, and fuel was never removed from the facility.
[f]Miscellaneous SNF from various reactors and sites now stored at ORNL.

repository. Additionally, separation of ^{233}U from waste material may be done to terminate safeguards requirements.

2.2.3 MSRE Material

Uranium-233 from the MSRE is the only unseparated material currently being recovered. This material consists of 31.01 kg of ^{233}U and 0.94 kg of ^{235}U. The uranium exists as UF_4 and is slowly being converted to UF_6 as the UF_4 reacts with radiolytically produced elemental fluorine from fluoride salts in the fuel. This UF_6 is being trapped on NaF pellets and shipped to Building 3019 for temporary storage. All of the MSRE salts are to be stabilized to an oxide form for long-term storage.

2.3 WASTE MATERIALS

About 85 kg of ^{233}U are in waste storage across the DOE complex. Like SNF, if this material is recovered, it could add a significant percentage to the inventory. This material exists in a wide variety of concentrations and retrievability. The threshold amount of ^{233}U that should be recovered is being determined.

Definitions for *waste material*, *exception-case material*, and *concentrated fissile material* have recently been proposed (Forsberg, Storch, and Lewis 1998). These definitions are proposed for current material with some basis on historical considerations. Recommendations have also been made for future or repackaged material to eliminate the exception case.

Waste material is defined as ^{233}U material that has no existing, planned, or proposed use. To be considered waste, the material must have an approximately homogenous concentration of <1 kg $^{233}U/m^3$ or have an enrichment level of <12 wt % ^{233}U in ^{238}U. Future waste material must also meet this definition.

Exception-case material is defined as material that should be examined on a case-by-case basis to determine if it is waste. This includes material, such as the INEEL RWMC rods and pellets, that do not meet the above definition of waste, but contain < 5 wt % ^{233}U (in non ^{238}U containing materials). Treatment of exception case materials depends on future decisions on criticality, safeguards, and arms control.

Concentrated fissile material is defined as all other ^{233}U-containing material, except for SNF. This material is the focus of the remainder of this report.

3. STORAGE OF ^{233}U

3.1 STORAGE REQUIREMENTS

Because of its unique characteristics, ^{233}U requires special handling and storage (Bereolos et al. 1998). The basic facility requirements for storage of any fissile materials are safeguards, criticality control, ventilation, and shielding. A specialized facility for ^{233}U is needed because of its differences from plutonium and HEU, especially with regard to ventilation and shielding, if necessary. Uranium-232 is almost always present with ^{233}U and has as part of its decay chain ^{208}Tl, which emits a highly penetrating 2.6-MeV gamma-ray accompanying its beta decay to stable ^{208}Pb. Because of this emission, ^{233}U requires special shielding and remote handling.

Ventilation is used as a means of physical confinement. In terms of alpha specific activity, ^{233}U is more active than HEU, but it is less active than most plutonium isotopes. However, ^{233}U also has a unique ventilation requirement imposed by the decay chain of its associated isotope, ^{232}U. Part of the ^{232}U decay chain includes ^{220}Rn, which normally exists as a gas and has a 55.6 s half-life. Thus, storage facilities for ^{233}U must consider the presence of this gas so that the radon is retained (for about 10 min, before final filtration) until it decays back to a particulate isotope that may be filtered.

Uranium-233 also differs from HEU and plutonium in terms of safeguards and criticality. The most important of these differences is more of a regulatory distinction than a technical one. Like HEU, the weapons usability of ^{233}U may be removed by isotopically diluting it with depleted uranium (DU). However, unlike HEU, there is no consensus on what constitutes weapons-usable ^{233}U. Thus, the International Atomic Energy Agency (IAEA) safeguards requirements have no distinctions for different isotopic purity levels of ^{233}U. Even material which has such a low isotopic composition of ^{233}U that it cannot reach criticality, let alone be used as a weapon, must be placed under safeguards measures appropriate for weapons-usable material. This makes disposal difficult and provides little incentive for isotopic dilution because there is no lowering of safeguards costs to compensate for the increased costs of handling and storing the larger mass of the diluted material.

3.2 CURRENT STORAGE SITES

The bulk of the unirradiated ^{233}U is stored at two sites, in Building 3019 at ORNL and at INEEL. The following sections describe the configurations, identify concerns, and address the vulnerabilities at these two sites.

3.2.1 ORNL Building 3019

3.2.1.1 Current Configuration

Building 3019 is a Manhattan Project era facility. It was built to perform the first plutonium separations from irradiated reactor fuel and to operate as a pilot plant for the Hanford facilities. Later, it was used to demonstrate other nuclear fuel processes on a pilot scale. The current mission of Building 3019 is to serve as the national repository for ^{233}U. As a part of this mission, Building 3019 still has the capability to process multikilogram quantities of ^{233}U.

Four sets of top-loaded, shielded, ventilated storage tube vaults are used for long-term storage. All four sets are accessible from the Penthouse (Room 201) of Building 3019 (Fig. 3.1). Each tube vault is vented through a manifold (called the Vessel Off-Gas (VOG) system) to the ORNL process off-gas system, which exhausts to Stack 3039. As backup ventilation, the VOG manifold branches to an independent system, the Building 3019 Cell Off-Gas (COG) system, which exhausts to Stack 3020.

One set, an array of 68 tube vaults, is installed in the southwestern corner of Cell 4, extending up into a 9- by 9-ft. former equipment hatch. These tube vaults are arranged in a triangular pattern, and each consists of a carbon-steel pipe, which is encased in a hexagonal, concrete structure (Fig. 3.2). The structure extends from the cell floor to about 1 ft. above the concrete hatch opening; thus, each pipe is about 32 ft. long with the top 6 ft. being a 6-in.-diam expanded section for shield plugs, ventilation connections and locking devices. The pipes inside 45 of the tube vaults are constructed from 4-in.-diam, Schedule 40, pipe, and the pipes inside the other 23 tube vaults are constructed from 5-in.-diam (outside), 0.25-in.-thick tubing.

Three sets of intercell (or in-wall) storage tube vaults (a total of 26 tube vaults) are located, respectively, in the shield walls separating Cells 2 and 3, Cells 3 and 4, and Cells 4 and 5 (Fig. 3.3). These tube vaults also are accessible from the Penthouse. Within each hole, a 4-in.-diam., Schedule 40, SS pipe serves as the storage tube vault. Each of these pipes is also vented through a manifold to the VOG System, and the top of each tube vault is shielded with an 8-in.-thick removable plug. The tube vaults between Cells 3 and 4 and Cells 4 and 5 are single rows of tube vaults about 3 in. from the center plane of the between-cell shield walls (avoiding a construction joint located in the center plane of the concrete wall that is equidistant from the cell interiors). The tube vaults between Cells 2 and 3 are oriented in a triangular pattern (in two rows); therefore, the concrete walls do not provide sufficient shielding. Thus, larger holes were drilled and lead shot was added to the annulus surrounding the pipe tubes to augment shielding.

All four sets of tube vaults are seismically qualified. Nuclear criticality safety is maintained by a combination of mass, geometry, and concentration controls. Limitation of neutron interaction with materials in adjacent tube vaults also maintains criticality safety.

Three pieces of equipment are used for material handling. A can-lift device provides a remote mechanical/electrical means of retrieving cans from the Cell 4 or the intercell tube vaults. Cans containing radioactive material are put into or retrieved from the storage tube vaults by one of several types of lifting or handling devices that are actuated by vacuum, electromagnet, or mechanical linkage (or a combination of actuators). These devices also can be used to install or remove a can into or out of a shielded transfer cask.

A 10-ton crane provides means for moving large pieces of equipment into, out of, or within the Penthouse. This crane also provides services (a) for accessing the Cell 4 and intercell storage tube vault and (b) for moving radioactive material containers into or out of the storage tube vaults. Shipping drums, shielded casks and equipment may be brought into or removed from the Penthouse via the adjacent crane bay using the 10-ton, bidirectional bridge crane. Such items may also be moved within the Penthouse with this crane.

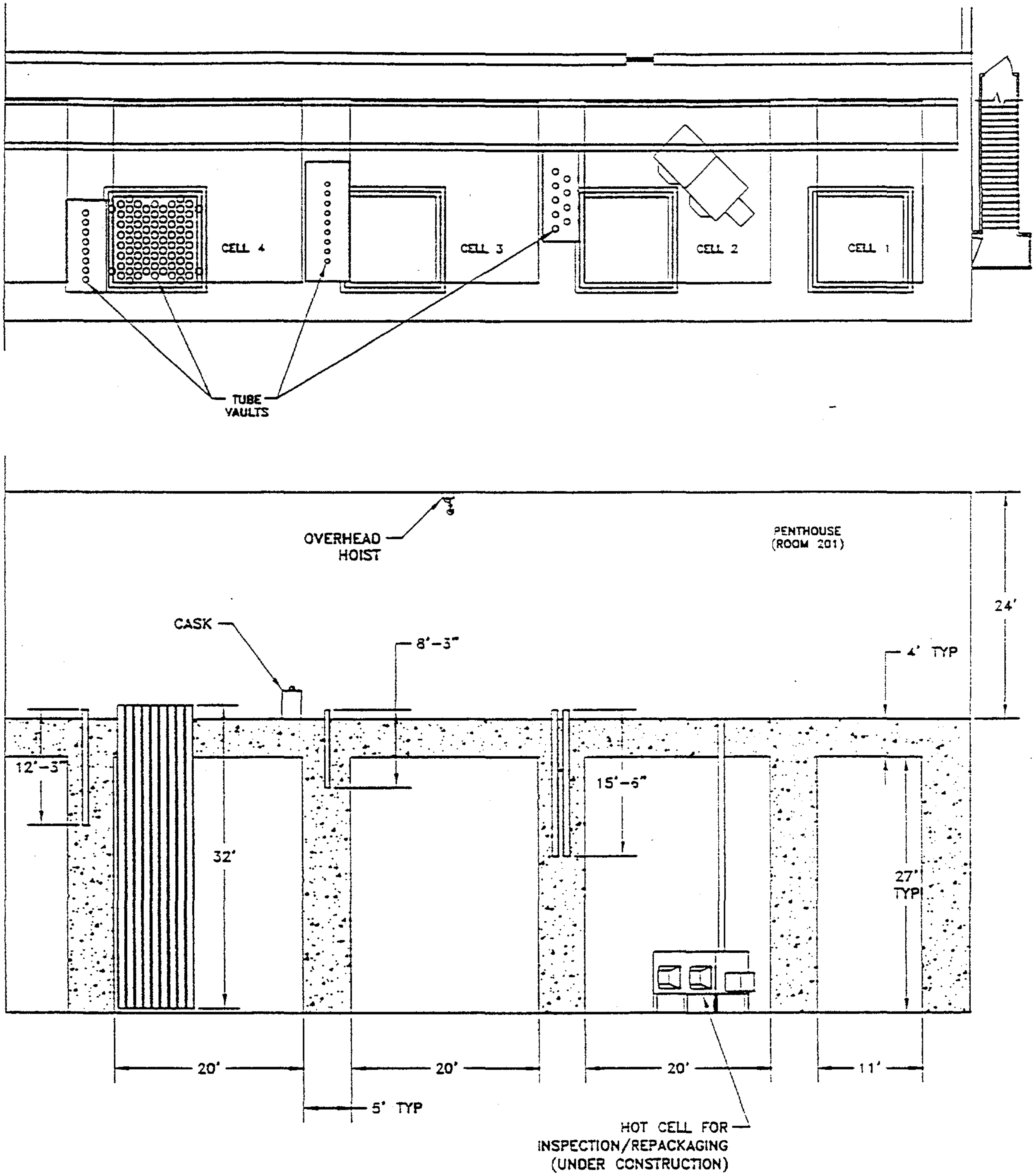

Fig. 3.1. Building 3019 Storage Configuration.

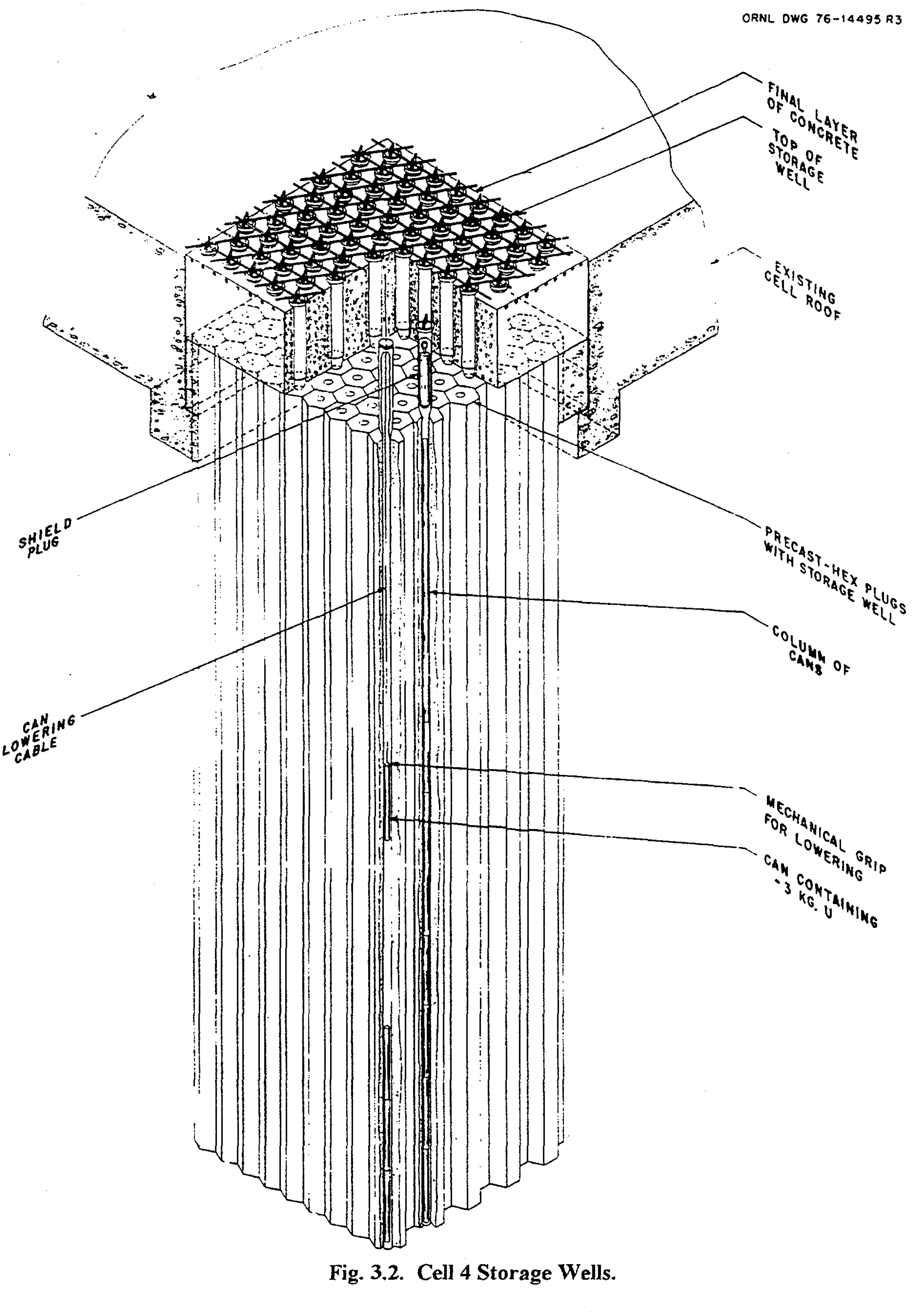

Fig. 3.2. Cell 4 Storage Wells.

ORNL DWG 98-4487

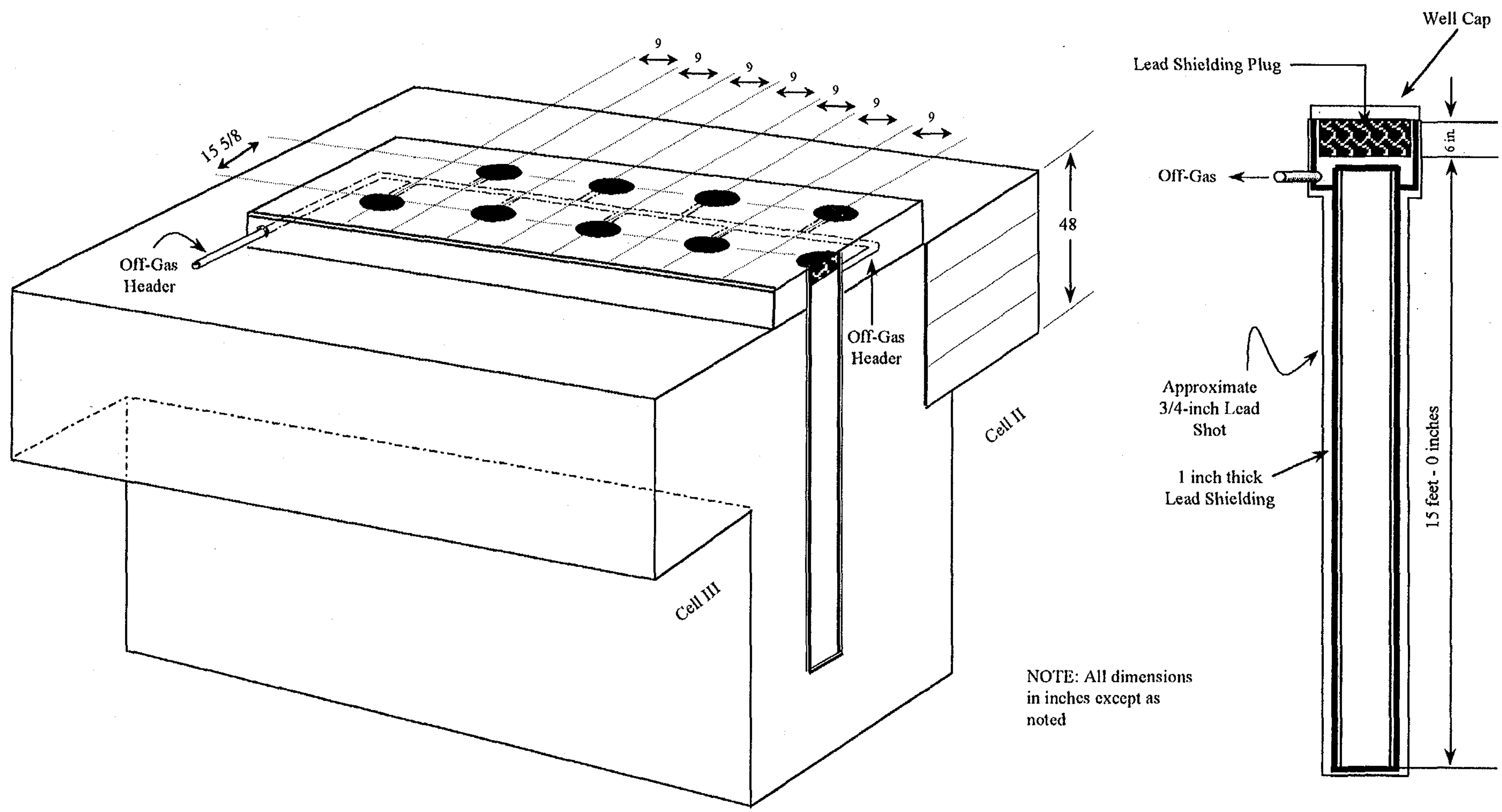

Fig. 3.3. Cells 2 and 3 storage tube vaults.

A shielded transfer cask provides a shielded and remote means of retrieving cans from the Cell 4 or intercell storage tube vaults. A shielded transfer cask may be used to move radioactive material containers with higher radiation levels than can be handled with shipping drums (ORNL Chemical Technology Division 1996*a*).

Although Category I levels (highest level of required safeguards) of fissile material are stored in Building 3019, passive security is used. The high radiation levels associated with some of the material provide a large degree of self-protection (> 100 R/hr on contact). Cameras, locks, and tamper-indicating devices provide supplemental security. Guards are used only when the material is being accessed.

3.2.1.2 Safety Concerns and Needed Upgrades

A potential safety concern is the possibility of a deflagration or explosion in a storage tube vault from the ignition of hydrogen that has evolved from radiolytic decomposition of hydrogen-containing materials (e.g., moisture or plastics). The likelihood of these occurring is dependent upon the tendency to trap hydrogen in a storage tube vault at concentrations above the lower explosion limit. This potential event also presumes that an ignition source is present in the hydrogen-trap volume (as might be plausible while removing locking devices to access stored cans) to initiate such a reaction. It is important to note that, currently, this scenario is considered a credible, but extremely unlikely, event that poses a marginal risk to on-site workers (0.5-25 rem at 240 meters) and potentially serious risk to the public (≥ 5 rem at site boundary). (More strict thresholds apply to members of the public than to on-site workers to arrive at these risk designations.) However, studies are underway to determine even the credibility of such a scenario for the configuration of the storage tube vaults and off-gas connections. Until these studies are complete, the Building 3019 Basis for Interim Operations (BIO) (Chemical Technology Division 1996*a*) recommends inert-gas purging of the tube vault and sampling of the tube vault head space during the tube vault access.

Another area of concern is the condition and design basis of the ventilation systems. Some sections are almost 40 years old, built to less-demanding standards, and in a deteriorated condition. Since the ventilation systems are a key means of maintaining material confinement, they need to be updated and kept in good condition. These systems were not originally identified as safety class systems, but the Glove-Box Off-Gas (GBOG) is currently a safety significant system. This means that hazard analyses have determined that it must function to mitigate the consequences to workers for credible accident scenarios.

Likewise, portions of the VOG are not safety systems. In its current configuration, the VOG exhausts through the 3039 stack. However, this stack is not seismically qualified and is not considered a nuclear facility, meaning it should handle only exhaust streams from nonnuclear facilities. From a safety viewpoint, this is an acceptable arrangement since the hazard analyses in the BIO take no credit for the VOG system. However, since Building 3019 is a nuclear facility, a better alternative would be for the VOG to exhaust to Stack 3020 along with the rest of the Building 3019 ventilation systems. To accomplish this and to keep the tube vaults (most contaminated area) at the greatest negative pressure, a new ventilation branch or system (tube vault off-gas) needs to be established. Portions of the existing ventilation system might be modified for use in the new system.

Numerous upgrades are currently being performed on or are planned for the Building 3019 ventilation systems. It is of great importance that a design baseline be established and configuration management be implemented for these changes. The interrelation of the various ventilation systems means that changes on one part of the system may cause significant changes to other parts. Some phases of a configuration management program have already begun. Current drawings are being confirmed for accuracy and cataloged in a database.

3.2.1.3 Vulnerabilities and Corrective Actions

The DOE *HEU Environmental, Safety, and Health (ES&H) Vulnerability Assessment* (U.S. DOE 1996*b*) identified six vulnerabilities in the 3019 Complex. This section lists the vulnerabilities and gives possible actions that can be used to correct the vulnerabilities.

- *Seismic and wind capacity not evaluated.* A generic vulnerability for the ORNL site that the seismic and wind capacity of many of the buildings have not been evaluated per current DOE requirements. For Building 3019, this vulnerability applies to the areas outside of the storage tube vaults. This vulnerability does not indicate a lack of qualification, only a lack of evaluation. A complete natural phenomenal hazards analysis for the Building 3019 complex is being performed in conjunction with the update of the Building 3019 Facility Authorization Basis (FAB).

- *Failure of high-efficiency particulate air (HEPA) equipment during a severe seismic or wind event.* Two other vulnerabilities deal with natural phenomena. One is failure of HEPA equipment during an earthquake or a tornado. This specifically pertains to the section of the VOG that remains above ground outside the southeast part of the building. This vulnerability could be dealt with during the seismic analysis of the areas outside the storage tube vaults. However, it is already being addressed in an active project that will harden this section of the VOG.

- *Failure of Tank P-24 due to a seismic event.* The final vulnerability from a natural event pertains to failure of the P-24 tank during an earthquake. The P-24 tank is located in a concrete bunker next to Building 3019 and stores uranium and thorium nitrate solutions. Again, an overall seismic evaluation of the Building 3019 complex may address this problem. If the P-24 tank does not meet the criteria in the DOE standards, then a decision must be made as to the future storage of this material. Sealing the material in grout would reduce the likelihood of release during a natural disaster. However, no schedule for such a plan has been made, to date, because the material is still useful as a neutron poison.

- *Leakage of thorium nitrate solution from Tank P-24.* Another vulnerability associated with the P-24 tank is the possibility of a spill during transfer of material. During the storage period of liquids in the P-24 tank, it may be necessary (e.g., because of a leak in the tank) to pump the entire inventory into an adjoining tank or even a nearby temporary tank. If the transfer were to be performed unattended and a leak in the line developed, the entire contents could be released to the environment while being pumped. However, an analysis of the transfer process has shown that it would take 14 h of pumping at the maximum flow rate before the minimum dose limit requiring corrective action would be reached (Webb 1996). Therefore, by monitoring transfers more frequently than once every 14 h, this accident scenario will be prevented. A procedure requirement for periodic monitoring during these transfers has eliminated this vulnerability.

- *Failure of ^{233}U can in storage tube vault.* The final two vulnerabilities deal with failure of cans of ^{233}U. The first of these is the possibility that a can has failed within a storage tube vault. This might result from corrosion during long periods of storage or by overpressurization resulting from radiation effects on the materials inside the can. Because of lack of inspection capabilities, most packages have not been removed since being placed in the tube vaults. The longest residence time is 33 y. The average is 15 y. A physical inspection of the material in the tube vaults has begun. The package conditions are being evaluated, compared to a storage standard, and repackaged, as required.

- *Failure of ^{233}U oxide can during handling.* The other vulnerability associated with cans of ^{233}U is the possibility that the aged or corroded can fails while being handled. The most likely scenario for such an event could occur if the can was dropped because of a failure in the can-lift device. This issue was examined in the Unreviewed Safety Question Determination (USQD) for the ^{233}U shipment from Mound Applied Technolgies in Ohio (ORNL Chemical Technology Division 1996*b*). Two separate scenarios were examined. In the first, a can of powder was dropped ~5 ft onto the floor of the Penthouse. In the other scenario, the can was dropped ~35 ft down a storage tube vault where it impacted the can(s) below it. Both cases were bounded by accidents analyzed in the Building 3019 BIO.

  Because there were no unresolved safety questions, this USQD is being incorporated into the BIO and should be applicable to any material consolidated from small-holdings sites. However, this analysis may not apply to material already located in the tube vaults because (a) the material examined in the USQD for the Mound material does not bound the material in some stored packages and (b) the condition of the cans was known to be good. For material already in the storage tube vaults, the container condition is unknown. Therefore, the damage factor (the fraction of material at risk that is released in an accident scenario) may be higher.

  For the physical inspection of containers currently in the tube vaults, the dropped container scenario will be addressed by confinement augmentation. This will consist of engineered systems that provide confinement of the material in case of a failed can, thus protecting workers and preventing release of material to the environment.

3.2.2 INEEL

3.2.2.1 Current Configuration

Uranium-233 is being stored currently at two sites at INEEL. Unirradiated fuel is stored at ICCP in underground storage vaults of the LWBR Fuel Storage Facility (CPP-749). Other materials are stored at the RWMC in two areas: the Transuranic Storage Area (TSA) and the waste storage buildings.

The unirradiated fuel is stored in underground dry storage vaults that are located at the INEEL-ICPP CPP-749 area. The CPP-749 area is an underground dry storage facility for reactor fuel. Each unirradiated fuel storage dry vault is a lined hole sized to hold two fuel storage canisters end to end and designed to isolate the fuel. A total of 21 dry vaults (labeled U-1 through U-21) were built to store forty 8-5/8-in.-diam unirradiated fuel storage canisters and to maintain 1 spare storage space. A 30-in.-diam dry vault (labeled U-22) contains the unirradiated seed module.

Each vault position in CPP-749 for storing the unirradiated LWBR fuel contains unirradiated fuel rods that are stored in SS containers or canisters. Each canister is constructed from 8-in.-schedule 80 SS pipe and is called a fuel-handling unit (FHU). The unirradiated fuel rods are stored in a total of 40 canisters that fill 20 vault positions. In addition, one spare unirradiated seed module, contained in a canister, is stored in an additional vault (U-22) in the unirradiated LWBR fuel storage area.

The TSA (Fig. 3.4) consists of ~100,000 drums of transuranic (TRU) waste on asphalt paving. Interspersed among the TRU waste drums are 107 drums of ^{233}U rods and pellets and ~1800 drums of ^{233}U waste. The drums are surrounded by an array of 4 by 8 waste boxes for structural support. Plywood decking is layered across the top of the drums. A plastic cover was placed over the plywood, and an earthen cover is mounded over the top and sides. The pad has now been covered by a steel building.

Drums containing LWBR rods and pellets are stored in 2R containers inside 6M shipping drums. Sixty-five of these drums were placed in shielded overpacks (6 drums/overpack) that are now stored on a concrete pad inside a RCRA-compliant storage building.

3.2.2.2 Vulnerabilities and Corrective Actions

The DOE HEU ES&H Vulnerability Assessment (U.S. DOE 1996b) identified five vulnerabilities relating to ^{233}U at INEEL. The details of these vulnerabilities and possible corrective actions are provided in this section.

- *Aging facilities with inactive quantities of HEU.* This is a generic vulnerability for the INEEL site. However, the two INEEL sites storing ^{233}U are not affected. CPP-749 was built in the early 1980s, and the TSA pad in the 1970s. Therefore, no corrective action is necessary.

- *Collocated and subsequent handling of $^{233}U/^{232}U$ as TRU waste at RWMC.* As the result of a DOE declaration, $^{233}U/^{232}U$ is being stored and handled as TRU waste. This collocation introduces a radiological hazard not normally associated with TRU waste.

- *Breach of ^{233}U drums collocated with TRU waste drums in Air Support Building-II* (ASB-II). The next three vulnerabilities concern corrosion of drums containing ^{233}U. Corrosion can compromise structural integrity and drum spacing. This creates a potential for an inadvertent criticality. The vulnerability of the containers stored in ASB-II has been corrected by inspecting the 12 drums that contain ^{233}U. These drums were then overpacked and moved to a RCRA-compliant storage facility.

- *Container Corrosion on ILTSF Pad.* Criticality safety on the ILTSF Pad was based on unverifiable space geometry. Corrosion of the drums could lead to compromise of container spacing and possible criticality. Fifty-three drums containing ^{233}U were inspected at ILTSF. The drums were then overpacked and relocated to the same RCRA approved storage facility as the drums from ASB-II.

- *Container Corrosion on TSA Pad.* The safety basis on the TSA pad is also based on spacing. Since the TSA pad is under earthen cover, the containers could be subject to corrosion and loss of integrity. However, the drums are protected from contact with the earth. Furthermore, packages recently removed from other TSA-like storage locations have not shown significant degradation. INEEL plans to remove this ^{233}U when the TSA is exhumed and the drums packaged for shipment to the Waste Isolation Pilot Plant (WIPP).

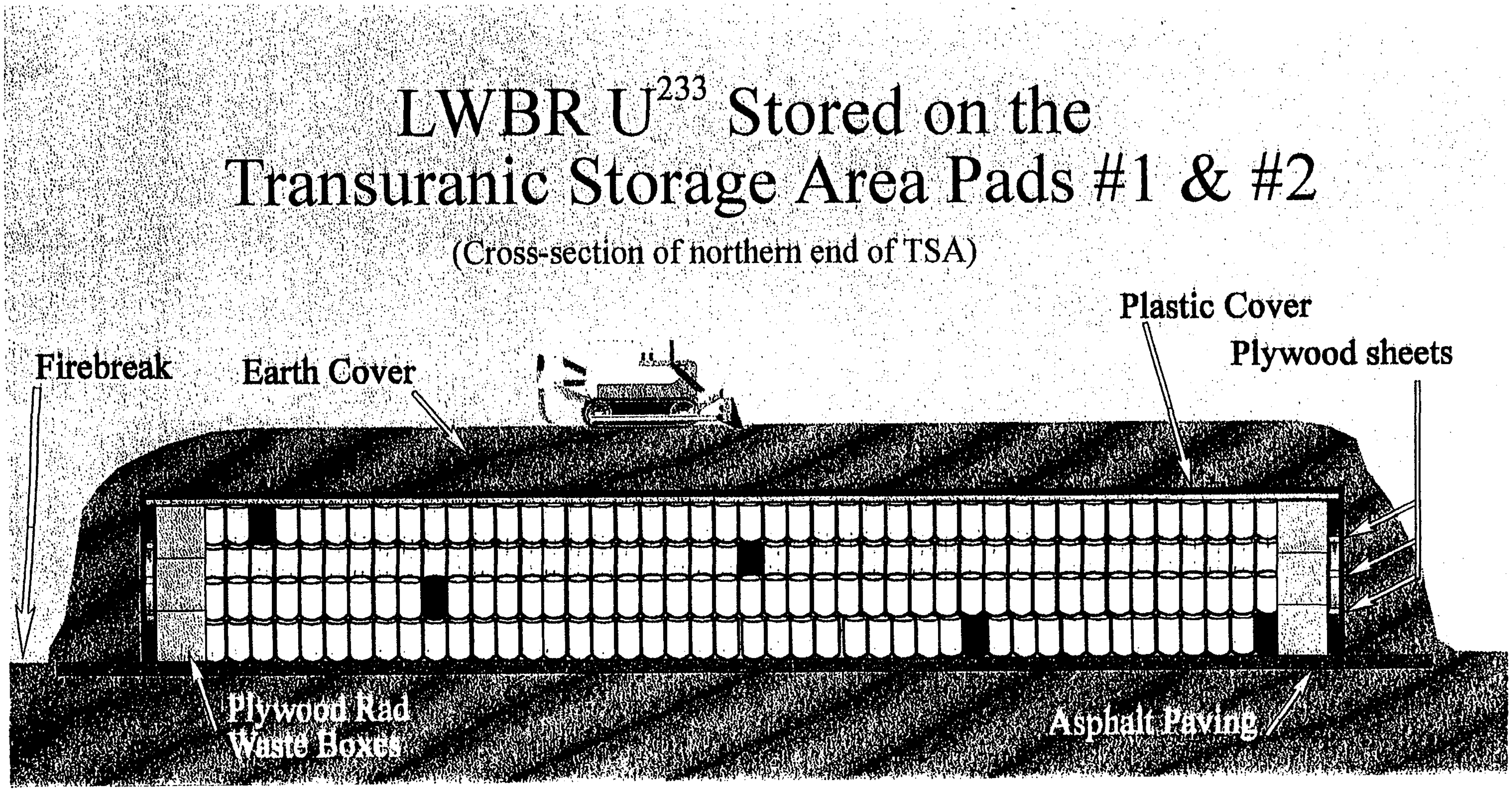

Fig. 3.4. Cross-section of the TSA asphalt storage pad at INEEL.

Courtesy of INEEL, Idaho Falls, Idaho.

3.2.3 Other Sites

Several additional sites have small holdings of ^{233}U (<10 kg/site). Only one vulnerability relating to ^{233}U was found for all the other sites: Los Alamos (< 1% of the total mass inventory) was storing ^{233}U in the TA-18 Hillside Vault, which has no HEPA filters on its exhaust stack [Defense Nuclear Facilities Safety Board (DNFSB), 1997]. Plans have been made to move the material in the Hillside Vault to ventilated storage.

3.3 STORAGE OPTIONS

Two policy issues are associated with storage of ^{233}U. The first is whether to store the material as high-isotopic-purity ^{233}U or to dilute it with sufficient ^{238}U to eliminate its potential use as weapons material. The second issue is whether to store the material in one or two centralized locations.

3.3.1 Isotopic Dilution

Isotopic dilution could eliminate many safeguards requirements and simplify storage. For many uses of ^{233}U, isotopic dilution would not eliminate its usefulness. Specifically, proposed medical applications would not be prohibited. While isotopic dilution would reduce security costs, it would, however, increase storage volume and processing costs. For other applications, high-purity ^{233}U may be required. An analysis of what fraction of the inventory could be isotopically diluted and remain useful has not been done.

In many respects, the policy issue concerning isotopic dilution of ^{233}U is analogous to the issue concerning the use of HEU as a research reactor fuel. For most, but not all, research reactor applications, HEU is not required. Similarly, for most, but not all uses, ^{233}U can be isotopically diluted to eliminate its potential as a weapons material without eliminating its usefulness.

The volume increase associated with isotopic dilution may require modifying existing ^{233}U storage facilities or building new ones. Alternatively, existing surplus buildings might be renovated.

3.3.2 Inventory Consolidation

ORNL is the national repository for separated ^{233}U and has its inventory in a variety of packages and diverse chemical and physical forms. INEEL-ICPP has a comparably sized inventory of ^{233}U in the forms of unirradiated nuclear fuel assemblies, rods, and pellets. Together these two sites hold over 98% of the current mass inventory. There are multiple small holdings of ^{233}U at other sites. These small inventories are primarily from earlier programs that no longer have a need for the material.

In terms of cost effectiveness, it makes sense to consolidate the inventory. Experienced personnel and the proper facilities are already in place at ORNL for storing a variety of ^{233}U packages. ORNL has already taken steps to consolidate the inventory from small-quantity sites. In 1996, the remaining inventory of ^{233}U at the Mound Plant was moved to ORNL. Discussions are currently taking place to bring ^{233}U materials at LANL and LLNL to ORNL. Capabilities are being put in place to actively handle, repackage, and chemically stabilize items posing higher radiation hazards.

A preliminary study has indicated that the cost of relocating the ^{233}U currently held in the repository would be on the order of 5–10 times the amount necessary to continue storing the material in an upgraded Building 3019 at ORNL (LaGrone 1991). Attempting to store ^{233}U in existing HEU vaults could also cause problems because of higher shielding and ventilation requirements. In the past, there have been problems with cross-contamination when ^{233}U and HEU were stored together (Uranium Storage Assessment Team 1996). A detailed study of the trade-offs between various alternatives for storing ^{233}U is currently underway as a part of the response to DNFSB Recommendation 97-1.

4. POTENTIAL FUTURE USES OF ^{233}U

Most of the current DOE inventory of ^{233}U is considered excess to national security needs. Some of this material will be disposed of as waste, but some will be kept for future use. Several potential uses of ^{233}U are described in this section. The decision as to which portions of the inventory should be retained for use depends on two considerations: (1) future needs and (2) existing inventory. Specific needs have specific requirements for the quality of ^{233}U. Not all ^{233}U is suitable for all uses. This may result in excess ^{233}U with specific characteristics, although the potential future uses of ^{233}U exceed the current inventory. The important characteristics of the inventory in terms of quality are:

- *^{232}U content.* Materials with high concentrations of ^{232}U are characterized by (a) high radiation levels, which impose restrictions on possible future uses, and (b) high processing costs to accommodate the radiation fields accompanying such materials.

- *^{235}U and ^{238}U content.* Materials with high concentrations of uranium isotopes other than ^{233}U indicate high volumes of uranium per unit of ^{233}U. If the other uranium isotope is ^{235}U, additional complications exist in terms of safeguards and nuclear criticality. If the other uranium isotope is ^{238}U, the material mass is increased, but weapons safeguards issues can be avoided, and nuclear criticality issues can be reduced if the fissile concentrations are sufficiently low.

- *Chemical and packaging characteristics.* The ^{233}U inventory is in multiple chemical and physical forms and packaging systems. Large fractions of material are cast-in-place oxide monoliths in welded SS containers; UO_2-ThO_2 ceramic pellets in Zircaloy or SS tubes; or oxide powders packaged in SS screw-top containers, welded aluminum cans, or welded SS plates. A variety of other chemical forms exist in other, diverse packaging configurations in the inventory. The diversity of chemical and physical material characteristics and packaging systems influences ^{233}U usefulness and complicates the approach to its use or disposition.

4.1 POTENTIAL MEDICAL APPLICATION

One potential large-scale use for ^{233}U involves one of its decay products, ^{213}Bi. Over the past decade, considerable research has been conducted in alpha-radioimmunotherapy. Specifically of interest is that of antitumor antibodies radiolabled with an alpha emitter (Knapp and Mirzadeh 1994; Geerlings 1993). In this method, the isotopes are attached to antibodies that specifically target cancer cells; the resulting alpha emissions kill these cells with high efficiency.

Previous work in this area focused on using ^{212}Bi, produced by the decay chain of ^{232}U (or ^{228}Th). However, the undesirable side affect of ^{212}Bi is the 2.6-MeV gamma radiation emitted during the decay of ^{208}Tl. The radiation level from this decay could prove to be a debilitating hazard to the patient and an unacceptable risk to the patient's family members and the medical staff involved in the treatment. There are also particular concerns about long-term dose levels to medical personnel who treat multiple patients.

A potential solution to this dilemma is to use ^{213}Bi produced from the decay chain of ^{233}U (Pippen 1995). Bismuth-213 has the unique properties of being primarily an alpha emitter (by way of ^{213}Po) and having only a 2% probability of decaying to ^{209}Tl, which emits a 1.5-MeV gamma-ray (this compares to a

36% probability for ^{212}Bi to decay to ^{208}Tl, which emits a 2.6 MeV gamma). Still, it is chemically identical to ^{212}Bi with a similar half-life.

Recovery of ^{213}Bi involves a three-step process (Fig. 4.1). First, ^{233}U is dissolved in acid and ^{229}Th and its decay products are separated from the uranium by ion exchange. The resulting thorium-bearing solution contains essentially no fissile uranium, has no nuclear weapons use, and, therefore, poses no complications in terms of safeguards or nuclear criticality. Next, ^{225}Ac is separated from ^{229}Th and the other decay products. Because actinium is not a part of the decay chain of ^{232}U, this separation removes the undesirable product ^{208}Tl and its precursors. Finally, a biomedical generator system may be loaded with ^{225}Ac, from which ^{213}Bi may be "milked".

After the first recovery step, the remaining uranium in solution is resolidified and stored in standardized packages for future use or disposal. The entire process may be repeated after several years to allow for ingrowth of ^{229}Th and other decay products. However, current plans are to separate ^{229}Th from ^{233}U once and then to disposition the ^{233}U.

Currently, ^{229}Th produced from the decay of ^{233}U is the only source of ^{213}Bi. Further ^{229}Th could be produced by irradiation of ^{226}Ra in a nuclear reactor. However, the existing capacity for such production is only about 100 g/year (Feinendegen and McClure 1996). Furthermore, the levels of the contaminant, ^{228}Th, produced by irradiation of Ra, are much higher than those from decay of ^{233}U/^{232}U in the inventory.

It is likely that isotopic dilution of the ^{233}U to remove weapons usability would have little effect on this application. The decay chain of ^{238}U, which would be used as the blend down material does not contain actinium. Therefore, the third separation step in the recovery of ^{213}Bi would still isolate the desired part of the ^{233}U decay chain. However, a much larger mass of material would have to be processed per gram of ^{213}Bi recovered.

4.2 FUEL IN NUCLEAR REACTORS

Direct use of ^{233}U for fabricating reactor fuel is possible for both DOE's research reactors and as part of a larger program to develop a mixed oxide (MOX) fuel of plutonium and uranium. Because of the limited inventory of ^{233}U, the latter option would not be an economic option for using ^{233}U by itself, but could be viable as a part of the plutonium program.

4.2.1 Deep-Space Missions

Because ^{233}U has a lower minimum critical mass than ^{235}U or ^{239}Pu (for neutron flux in the thermal regime), it may be desirable to use it as a nuclear reactor fuel for deep-space missions, for which a premium is placed on minimizing mass. For this application, high-grade ^{233}U would be used to minimize spacecraft launch weight. A space reactor is put into earth orbit before it is started. This avoids the need for massive shielding of the reactor before and during launch operations. High-purity ^{233}U would be required to avoid the need to shield the reactor before launch.

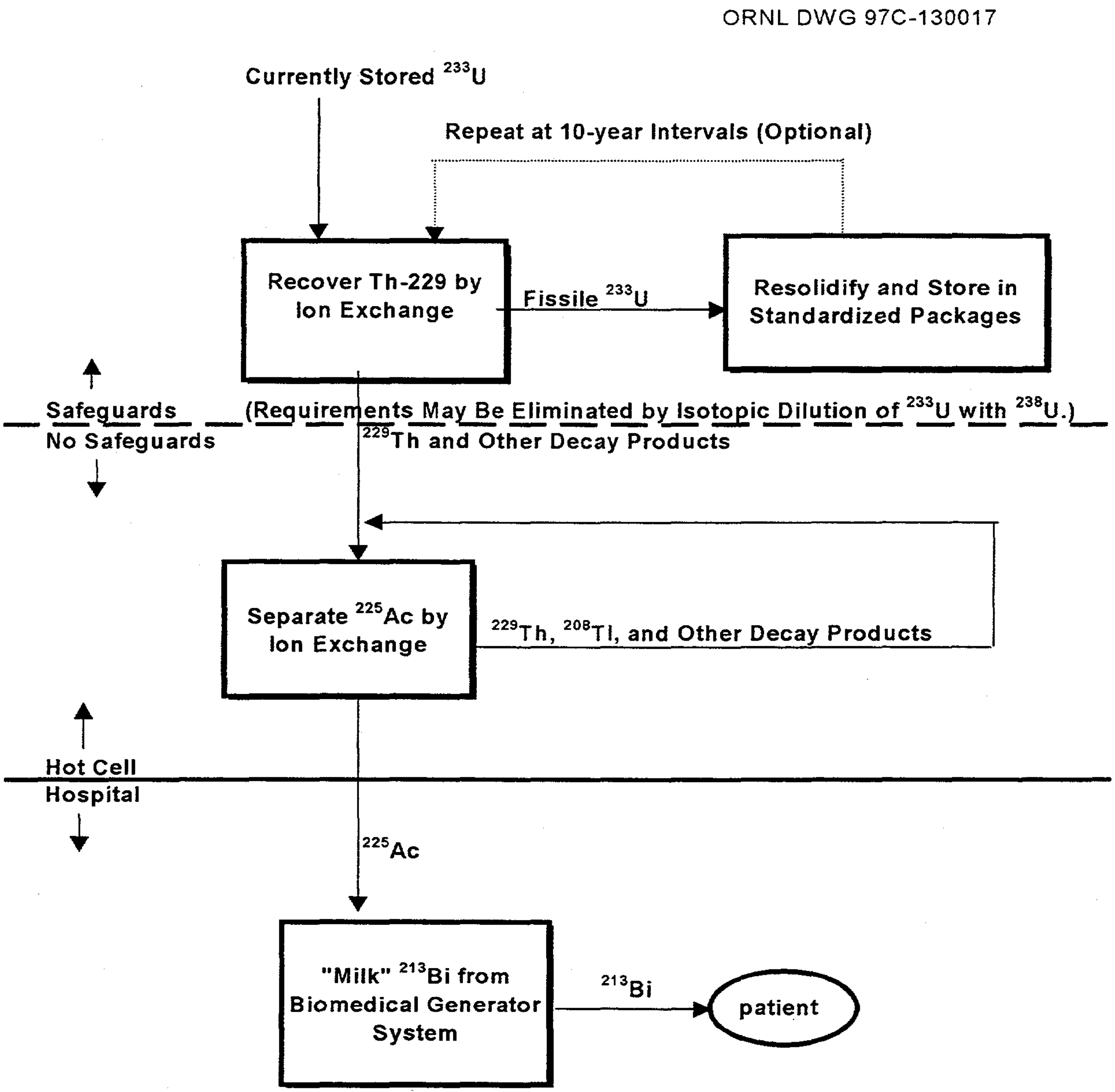

Fig. 4.1. Flowsheet for ^{213}Bi production.

Uranium-233 may have unique advantages for certain types of deep-space reactors (Howe 1991). There are two potential applications: (1) electric power and (2) propulsion. However, there are two constraints for deep-space missions. For missions to Mars and beyond, nuclear energy sources are the only available sources of electric power for spacecraft because of the reduction of solar radiation with distance from the sun. Second, the cost of deep-space missions is directly dependent upon the total weight of the spacecraft. While the cost to put satellites in earth orbit is measured in thousands of dollars per kilogram in orbit, the cost to put a spacecraft beyond Mars may be measured in millions of dollars per kilogram. Thus, weight controls costs.

The preferred type of nuclear power source to provide electricity for a deep-space mission depends upon the power requirements. For power production levels up to several kilowatts, the minimum-mass nuclear power source is a radioisotope generator. The currently preferred radioisotope is ^{238}Pu. Nuclear reactors provide minimum-mass, steady-state power generation at higher power levels. For steady-state power levels of a few kilowatts to several megawatts, nuclear power reactors fueled with ^{233}U may provide the minimum mass (MacFarlane 1963; Lantz and Mayo 1972). For each fissile material, a minimum mass of that fissile material is required for a nuclear reactor to operate. This minimum mass is substantially smaller for ^{233}U than for ^{235}U. Uranium-233 and plutonium have similar nuclear characteristics; however, the physical properties of uranium in high-temperature space reactors are substantially better than those of plutonium, and there may be fewer launch safety issues. This may make ^{233}U the preferred material for such applications.

At steady-state power generation levels of several megawatts and above, ^{233}U, HEU or plutonium each can be used with little difference in spacecraft weight. There are two reasons for this:

- At high-power levels, the reactor must have large internal heat-transfer surfaces to transfer heat from the reactor to the electric generator. The reactor fuel assemblies to obtain the heat transfer require a significant amount of fissile material. In a large nuclear system, the choice of fissile material does not significantly impact weight because the amount of fissile material needed for heat transfer purposes far exceeds the minimum critical mass needed for a reactor.

- At high-power levels, there must be significant quantities of fissile materials to provide the energy for a long-term mission.

Uranium-233 may also be used for small nuclear propulsion units to boost spacecraft from earth orbit to deep space (Ludewig et al. 1989; Hyland 1970). These units have moderate-power levels for short times (<1 h). The interest in using ^{233}U is that it minimizes weight.

It currently appears unlikely that ^{233}U would be used for near-earth missions because of the interactions between safety concerns and economics. The primary safety issue associated with space reactors is a launch failure with loss of the spacecraft. Space reactors are not operated until they are in earth orbit or beyond to minimize launch risks. For reactors fueled with HEU, launch safety concerns are minimized because a failure of the rocket would result in only HEU dust over the launch pad or burnup in the atmosphere. The toxicity of HEU is relatively low and less than other components of conventional rockets. Plutonium-238, ^{233}U, and ^{239}Pu are alpha emitters that are much more hazardous than is HEU. Uranium-233 is hazardous, but the least hazardous of these materials. For more hazardous radionuclides, the standard procedure is to encase the fissile material in a special container or containers to withstand launch accidents. This adds costs and complexities to the reactor. For a near-earth spacecraft, the relatively low-launch cost for earth orbit makes HEU the preferred reactor material — it is not worth the complexity of using ^{233}U or plutonium. For deep-space missions with very-high costs to deliver a kilogram of spacecraft into deep space, more complex systems are required to (1) make the mission possible and (2) limit costs.

The potential demand for ^{233}U in this application is probably a fraction of the total inventory. The unique advantages of ^{233}U are only for small reactors with small quantities of ^{233}U per reactor. Any ^{233}U for space missions must be of high purity. Uranium-233 with high levels of ^{232}U or with significant quantities of other uranium isotopes has no value for this application.

4.2.2 Reactor Fuel Cycle Research

The major historical application for ^{233}U has been for research into new nuclear power reactors and associated fuel cycles. This is also a potential future application.

There are four incentives for considering a ^{233}U-thorium fuel cycle.

- The global resources of thorium are about four times greater than those of uranium. If uranium becomes scarce, thorium is a more abundant fertile material to use in reactors to breed nuclear fuels.
- In thermal reactors, such as LWRs, thorium fuel cycles breed more fissile material (^{233}U) than reactors fueled with low-enriched uranium.
- SNF and other wastes from the thorium-^{233}U fuel cycle, compared to uranium-plutonium fuel cycles, contain far smaller quantities of long-lived actinides that are a concern in wastes to be disposed in geological repositories.
- Some ^{233}U-thorium fuel cycles have significantly lower risks of diversion of weapons-usable material than conventional uranium-plutonium fuel cycles. In power reactors, the impurity ^{232}U and its daughter products build up to very high levels with correspondingly high radiation levels associated with the separated ^{233}U.

For this application, only relatively pure ^{233}U would be used. For research, high-purity material with low radiation levels is desired to allow (1) low-cost fabrication of test nuclear fuel assemblies and other equipment and (2) make possible more accurate measurements of equipment and material performance.

The U.S. inventory of separated ^{233}U partly reflects the use of ^{233}U for power reactor research and development (R&D). The CEUSP material is from a power reactor. The zero-power reactor ^{233}U is from reactor criticality experiments. The Idaho LWBR fuel is a test core of ^{233}U.

A recent proposal that would involve the use of ^{233}U as nuclear fuel is the Energy Amplifier concept of Rubbia (Carminati 1993; Aldhous 1993; Rubbia 1995). The idea is to use a particle accelerator to supply neutrons to drive a thorium-fueled reactor. This setup has the safety advantage that fission can be sustained only while the accelerator is running. Therefore, a runaway nuclear accident is highly unlikely. It is also noted that Japan is currently acquiring limited quantities of ^{233}U for tests for its advanced reactor concepts.

The total inventory of ^{233}U (< 2 metric tons) is small compared to inventories of other fissile materials (HEU inventories are measured in hundreds of metric tons) and small compared to the fissile requirements to fuel the nation's nuclear power reactors. As such, the inventory is not a significant energy resource from a national perspective or a significant energy resource for start-up of a ^{233}U fuel cycle.

Decisions on keeping such materials for this application depend upon whether the US wants to maintain its ability to restart research on fuel cycles with lower proliferation risks than uranium fuel cycles. The nonproliferation characteristics are perhaps the most unique characteristics of the thorium-^{233}U fuel cycles

4.2.3 Commercial Reactors

The inventory of ^{233}U is not significant as an energy source. Current commercial reactor fuel is based solely on ^{235}U, so all licensing and specifications are set up for this situation. Furthermore, there is currently an excess of ^{235}U. The CEUSP material, which has large quantities of ^{235}U, would still have to undergo additional processing to remove the soluble neutron poisons cadmium and gadolinium.

4.3 NUCLEAR WEAPONS

Because ^{233}U is fissile, it has the potential to be used in nuclear weapons. The fact that it has a smaller critical mass than ^{235}U makes ^{233}U an excellent primary fissile material. In this respect, ^{233}U is superior to plutonium. However, the unavoidable presence of ^{232}U in ^{233}U material leads to complications because of strong gamma emissions. This is a serious safety concern in handling weapons made with ^{233}U. Because of the high exposure hazard, the radiation field around weapons constructed of ^{233}U with more than 10 ppm ^{232}U provides a degree of self-protection. Furthermore, the distinct gamma signature allows these weapons to be detected and tracked from a distance (Sublette 1997).

Any ^{233}U preserved for use in weapons programs would need to be kept in undiluted (weapons-usable) form. Uranium-233 is also used in small quantities as a diagnostic in tests of HEU weapons. If weapons tests resume in the future, there will be a small need for ^{233}U.

4.4 SPIKE MATERIALS

The ^{233}U isotope is used as a calibration spike in the determination of uranium concentrations and isotopic compositions in materials containing natural uranium or uranium enriched in ^{235}U. A spike is a measured quantity of an isotope that is added to an aliquot of a sample. The isotope used for spiking must either not be present or present only at trace levels in the original sample. In reprocessing spent uranium fuels, ^{233}U is not an actinide activation product or a fission product. After isotopic equilibration, the quantities of the isotopes in the sample are measured relative to the added isotope. From the change in the isotopic ratios of the sample relative to the spike, measured by mass spectrometry, the isotopic content of the sample may be very accurately determined. As a spike, ^{233}U has been used to:

- accurately measure the half-lives of other radioactive actinides, most notably ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu (Chitambar 1986; Abernathey and Marsh 1981; Jaffey et al. 1978);

- determine the concentration and isotopic composition of uranium in environmental air filters (Russ and Bazan 1997); and

- determine the concentrations of uranium, plutonium, and their relative isotopic abundances by an analytical technique known as isotope dilution mass spectrometry (IDMS).

IDMS is an effective method for accurately measuring element or isotopic assays and concentrations. It is especially useful when only small samples of an element or isotope of interest are available. IDMS is frequently used to accurately measure concentrations of uranium and plutonium in dissolved, irradiated nuclear materials such as those resulting from nuclear fuel reprocessing. As part of the IDMS analysis procedure, a known quantity of a unique element or isotope to be measured (referred to as the "spike") is added to a solution containing the analyte. The resulting solution is then chemically purified and then analyzed by mass spectrometry. By measuring the magnitude of the response for each isotope (including that for the unique spike) and then relating these results to the known quantity of the spike, the isotopic composition of the nuclear material under investigation can be accurately determined (Bayne 1991; Maxwell and Clark 1990).

For safeguards and accountability, concentrations of uranium and plutonium in highly radioactive solutions from dissolved spent reactor fuel elements are determined by IDMS using ^{233}U and other radionuclides as spikes. Such an analysis is also useful for determining the burnup level of reactor fuel (IAEA 1989).

Currently, ^{233}U is used as the spike material to calibrate the samples used for uranium accountability analyses that are performed in the ICPP at INEEL (Lockheed Martin Idaho Technologies 1995). A precisely measured aliquot (on the order of 1 mg) of an isotopically pure ^{233}U compound is added to samples to determine the concentration of uranium and of the accountable uranium isotopes present (mainly for the isotope ^{235}U) by IDMS. The analyzed sample size that is spiked with ^{233}U typically has on the order of one milligram of uranium. Discussions with INEEL-ICPP personnel involved with such analyses indicate that only about 20 g of ^{233}U have been used as a spike in IDMS analyses performed at the ICPP during the past 20 years (Hand 1997; Lewis 1997).

At the SRS, an automated spike preparation system for IDMS was developed. To prepare ^{233}U spikes for this system, 200 μL containing about 140 μg of ^{233}U is dispensed (Maxwell and Clark 1990).

4.5 RADIOACTIVE TRACERS

In general, tracers are foreign materials that are either mixed with, or attached to, a given substance and used to determine the location or path of that substance. Radioactive tracers are often used to follow a series of processes or events commonly found in industrial process streams or in the metabolic systems of living organisms. As a radioactive tracer, ^{233}U has been used in small amounts (~20 ng) to:

- track radionuclide migration in ground-water and other aquifer systems, (Laul 1985; Meier 1992; Meier 1994; Zeh 1995) and in geologic media (Shihomatsu 1987; Shihomatsu 1988),

- determine the concentration and distribution of uranium in minerals (Shihomatsu and Iyer 1988) and measure the diffusion of radionuclides in sediment rocks (Meier et al. 1987),

- calibrate other radionuclide tracers, such as ^{243}Am. In such calibrations, ^{233}U enables the evaporation rate of the test solution to be monitored (Eliot, Louis, and Lucas 1987).

4.6 OTHER (MISCELLANEOUS) APPLICATIONS

Other less common uses for ^{233}U materials have included:

- as a sensing material in fission counters for reactor startup applications (Prasad and Balagi 1996) (about 75 mg of ^{233}U are required in a fission counter providing a sensitivity of 0.02 cps/nv), and

- one of several selected reference radionuclides used to measure the effects and limits of radiation exposure on embryo and fetal tissue (Matsusaka 1993). Such analyses have included investigations made of the deposition, distribution, retention, and toxicity of several radionuclides in prenatal and juvenile mammals (Sikov and Park 1987).

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Appendix A. ^{233}U INVENTORIES

A.1 INTRODUCTION

This appendix lists current inventories of ^{233}U at DOE and commercial sites. Information on the inventories reported is based on several sources, mainly site data submittals. One of these information references (Davis Apr. 26, 1996) includes an unclassified listing of sites reported on the output received from a recent query made of the Nuclear Materials Management and Safeguards System (NMMSS) database. NMMSS is currently maintained by Nuclear Assurance Corporation (NAC) International and can be queried by contacting and making special arrangements with the NAC NMMSS Group in Norcross, Georgia (phone: 770/662-8110, Ext. 14). It should be noted that the NMMSS database, as currently maintained by NAC International, is classified and that this appendix reports unclassified ^{233}U inventories. Currently, the ^{233}U Disposition Program at Oak Ridge National Laboratory (ORNL) is directing efforts toward declassifying additional information and data associated with site ^{233}U inventories to facilitate the assessment and evaluation of various ^{233}U material disposition options.

In general, this report does not include inventories of ^{233}U from spent nuclear fuel (SNF). Materials that are considered part of the ^{233}U inventory include unirradiated pelletized reactor fuel, salts, uranium alloys and metals, uranium oxide (UO_x) powders, U_3O_8 monolithic materials, and miscellaneous uranium compounds and materials in irregular forms.

Figure A.1 is a map showing current locations and the magnitude of stored, accountable ^{233}U mass inventories at each of the 13 DOE and 3 U.S. Nuclear Regulatory Commission (NRC)-licensed sites listed in Table A.1.

A.2 INVENTORIES

Table A.2 summarizes the current inventories of ^{233}U at each domestic site. The information and data reported in certain columns of this table are based on specific code definitions defined in Tables A.3, A.4, and A.5. Table A.2 lists the following ^{233}U information and data:

- material form description (based on the material type and form listed in Table A.3),
- packaging types (based on the packaging types and codes listed in Table A.4),
- number of packages,
- material type code (based on the codes listed for ^{233}U and ^{235}U in Table A.5), and
- accountable mass (by total U, ^{233}U, and ^{235}U).

Tables A.6 through A.21 describe the separate materials at each site in more detail. Some of the site information and data in these tables were received as a result of the recent U.S. Department of Energy (DOE) vulnerability assessment made of highly enriched uranium currently in storage [O'Leary (Feb. 22, 1996)]. The information and data presented in these tables (derived from the references listed in Sect. A.4 of this appendix) are useful in planning for the final disposition of these stored ^{233}U materials.

Most of the domestic ^{233}U inventory is found in two major types of materials: unirradiated Shippingport light-water breeder reactor (LWBR) fuel at Idaho National Engineering and Environmental Laboratory (INEEL)–

Idaho Chemical Processing Plant (ICPP) and Consolidated Edison Uranium Solidification Program (CEUSP) material at ORNL. Characteristics of the unirradiated Shippingport LWBR fuel are given in Table A.22, and those of the CEUSP material are given in Table A.23. Further details of the unirradiated LWBR fuel are given in Appendix B. Figure A.2 shows the design specifications for a storage can containing CEUSP solidified waste.

Table A.24 lists (a) the individuals at each site who provided the ^{233}U information for NMMSS and (b) the individuals or organizations who currently have responsibility for the management of the ^{233}U site material.

A.3 PLANNED MATERIAL MANAGEMENT ACTIVITIES

A summary of the future management of ^{233}U materials is given below on a site-by-site basis. This information is based on the site submittals received for this report.

A.3.1 Argonne National Laboratory–East (ANL–E)

Characteristics of ^{233}U materials at ANL–E are given in Table A.6, which is based on information and data reported in Lambert (July 11, 1996) and Barkalow and Poupa (Aug. 12, 1996).

The foil discs and metal discs identified in Table A.6 have been declared as surplus and approved for disposition as transuranic (TRU) waste. Currently, there is no place to ship such TRU waste from ANL–E. An interim TRU waste storage facility is being readied to replace storage bunkers scheduled for decontamination and decommissioning (D&D). The foil discs and metal discs are being held in a special materials storage vault pending removal by the site waste management organization.

The solution is in active use as a "spike" source for chemical research. The small quantities used in the "spikes" are removed and treated as waste. There is only 0.8 g of ^{233}U material involved with this solution.

The reactor fuel materials are being held as samples remaining from prior research. Currently, no disposition for these materials is planned.

A.3.2 Argonne National Laboratory–West (ANL–W)

ANL–W ^{233}U materials are indicated in Table A.7, which is based on Meppen (Aug. 21, 1996). Most of the items listed in Table A.7 are used as standards. Some are expected to be disposed of as waste, and only a small number of items are considered to be surplus.

A.3.3 Bettis Atomic Power Laboratory (BAPL)

The ^{233}U material at BAPL, reported in Table A.8 [based on Johnson (Sept. 17, 1996) as amended by Stevenson (June 12,1998)], is in the form of waste. This waste has ^{233}U entrained in rags and other items such as empty fuel cans, trays, etc. The two waste disposal sites to which waste material is shipped from Bettis are the Savannah River Site (SRS) and the Hanford Site. Based upon preliminary calculations, at least 90 wt % of the waste material is definitely TRU and the other 10 wt % is most likely TRU. The SRS does not consider ^{233}U to be TRU; however, the waste acceptance limit for ^{233}U is restrictive. Hanford considers ^{233}U to be TRU and controls it at a concentration-based standard that is restrictive. In addition, there is methylene chloride, a hazardous chemical, in some of the waste. In the state of Pennsylvania, this waste material can be designated as nonhazardous based on process knowledge of the means by which the waste was generated. The SRS would call this mixed waste because of a different interpretation of the U.S. Environmental Protection Agency (EPA) hazardous rules. Hanford would consider this material nonhazardous. Both Hanford and SRS cannot accept this material because of the ^{233}U content of the waste. The material will remain at BAPL until a facility becomes available that is capable of handling waste that is TRU or has a high ^{233}U content.

A.3.4 Commonwealth Edison Company, Dresden Reactor–Unit 3 (COMED3)

The COMED3 material listed in Table A.9 (based on Larson Sept. 18, 1996) is irradiated fuel rods. DOE will dispose of these rods in a repository for spent nuclear fuel (SNF).

A.3.5 Fort St. Vrain Reactor (FSVR)

As indicated in Table A.10 [based on Borst (July 2, 1996)], all ^{233}U inventories at FSVR are found in SNF. DOE has assumed ownership of all irradiated FSVR fuel described in Table A.10 and will determine the final disposition of that fuel.

A.3.6 General Atomics (GA)

All nonirradiated ^{233}U materials at GA are currently held under a closed DOE program and are considered surplus. These materials contain very small quantities of ^{233}U and include coated particles and a dosimeter. Characteristics of these materials are described in Table A.11 [based on Wisham (Aug. 21, 1996, and May 12, 1997)]. Recently, the DOE Oakland Operations Office indicated that both the coated particles and dosimeter can be shipped off site and disposed of as a waste. However, the dosimeter also contains 0.02 g of ^{239}Pu, which may not be acceptable for land burial.

A.3.7 Hanford Site (Hanford)

There is no planned disposition for the ^{233}U materials listed for Hanford in Table A.12 [based on Serier (Aug. 21, 1996)]. All of these materials are considered surplus and exclude ^{233}U materials stored at the Pacific Northwest National Laboratory (PNNL) (see Sect. A.3.13). Only the pure oxide material packaged in tinned food-pack (C0) would likely be shipped to ORNL.

A.3.8 INEEL–ICPP

The Peach Bottom Fort St. Vrain and LWBR material will remain in first-generation dry-storage silos. Only a portion of the LWBR material (from Shippingport) is unirradiated, and a small portion of that is stored in the Radioactive Waste Management Complex (RWMC). Detailed characteristics of the unirradiated LWBR fuel are given in Table A.22 [based on ORNL Chemical Technology Division (Sept. 22, 1995); DOE Idaho Operations Office (July 19, 1995); Laible (Aug. 1, 1996); and Bright (Nov. 12, 1996)] and are discussed further in Appendix B. The Bettis scrap will be removed from wet storage and conditioned for placement into a dry-storage configuration. Most miscellaneous laboratory materials are to be retained as standards.

A.3.9 Lawrence Livermore National Laboratory (LLNL)

Characteristics of ^{233}U materials stored at LLNL are given in Table A.14 [based on Maslin (Aug. 29, 1996)]. The current plan for the consolidation of these materials is to ship most of them to ORNL. Only one encapsulated item will remain at LLNL.

A.3.10 Los Alamos National laboratory (LANL)

Table A.15 lists a summary of the characteristics of unclassified ^{233}U materials at LANL. The information and data reported for LANL in this section are based on Martinez (Mar. 4, 1997, and July 9, 1997); Roth (Aug. 30, 1996, and Oct. 4, 1996). The inventory reported includes contributions from 135 items in 109 packages.

All ^{233}U material items that are active or planned programmatic use (PPU), as defined by DOE's Office of Nuclear Weapons Management (DP-22), will remain at LANL for experimental use. These items are not considered surplus. Should such items be beyond the scope of current or foreseeable experiments and/or be beyond the scope of recycle and recovery efforts, they would become either available for use at other sites or shipped to ORNL for storage.

LANL currently has 38 nonsurplus items designated either as active or PPU that must also satisfy national security requirements (NSRs). LANL has requested that the DOE-Albuquerque Office's Weapons Quality Division (DOE-AL-WQD) provide guidance on what steps need to be taken for these items to meet NSR.

Items designated neither active nor PPU are in scrap forms, which may be discarded or packaged for long-term storage. If discard authorization is obtained, these items are sent to the LANL Waste Management Facility for storage as waste. If packaged for long-term storage, the container used will be one based on the specifications given in the DOE *Storage Standards Document*, STD-3013 [based on U.S. DOE (Sept. 1996)].

Multiple actinide items from LANL may not be acceptable for storage at ORNL. Pending DOE-AL-WQD guidance, these items may be discarded.

A.3.11 Mound Applied Technologies (Mound)

As indicated in Table A.16 [based on Stadler (Sept. 16, 1996)], 3.5 kg of excess ^{233}U material resided at Mound as of the end of 1995. At that time, a signed Shipper Receiver Agreement for this material was in effect between ORNL and EG&G Mound. This excess material was shipped to ORNL in 1996 and is now included in the ^{233}U inventory for ORNL.

A.3.12 ORNL

The largest inventory of ^{233}U is found at ORNL. Detailed characteristics of the materials comprising this inventory are given in Table A.17 [based on Krichinsky (Aug. 25, 1997)]. Most of the ^{233}U material at ORNL is a result of CEUSP. Detailed characteristics of the CEUSP material are given in Table A.23 (based on Peer Consultants Dec. 23, 1987). The ORNL inventory also includes 20 packages of ^{233}U material shipped from Mound in 1996.

Uranium-233 is considered a surplus material to its owner, the DOE Office of Defense Programs. However, interest has been expressed in recovering the daughters of ^{233}U for the beneficial use of alpha radioimmunotherapy to fight blood-borne cancers through radiotagging cancer-specific monoclonal antibodies. The attractiveness of using ^{233}U for this beneficial purpose is a function of the material's isotopic quality (i.e., relative to the concentration of its contaminating sister isotope, ^{232}U). Higher-quality ^{233}U (containing nominally less than 50 parts ^{232}U per million parts ^{233}U, ppm) is more appealing for this use since it contains less ^{232}U and its associated penetrating gamma-emitting daughters. Lower-quality ^{233}U (containing greater than 50 ppm of ^{232}U) is less appealing and a definite candidate for declaration as a waste. However, if an aqueous process is used to isotopically dilute the low-quality ^{233}U (for nuclear criticality safety) in preparation for its disposal as a waste, it would be worthwhile to recover even the daughters of this ^{233}U before material disposal.

A.3.13 Pacific Northwest National Laboratory (PNNL)

PNNL has only 48 g of ^{233}U, and the characteristics of this inventory are given in Table A.18 [data from Andre (May 2, 1997)]. Four packaged items contain a total of 27 g of ^{233}U for programmatic use. These items are used as stock solutions and as analytical standards at PNNL. Eleven packaged items contain ^{233}U that is excess to current needs. Disposition of the excess items typically includes the option of waste disposal and will be made in accordance with instructions from DOE Richland Operations.

A.3.14 Rocky Flats Environmental Test Site (RFETS)

As reported in James (March 10, 1997), the ^{233}U inventory at RFETS, reported in Table A.19, is very small. This inventory consists of small quantities of combustible and noncombustible materials. The latter include light (nonfissile) metals. Because of the small quantity, it is expected that the ^{233}U material at RFETS will be handled eventually as waste and shipped to an approved waste disposal site.

A.3.15 SRS

As indicated in Table A.20 [based on Severynse (Sept. 4, 1996) and Krupta (Mar. 11, 1997)], the total inventory of ^{233}U at SRS is contained in irradiated SNF assemblies and 17 Mark 50A ^{232}Th target slugs containing ^{233}U, which are stored in the SRS reactor basins. The plan for the disposition of thorium targets at SRS is to dissolve the targets in the H Canyon, dilute with depleted uranium (in the canyon) to satisfy criticality concerns, and drop the solution into the SRS high-level waste tanks. The diluted ^{233}U will then be processed by the SRS tank farm system, with final dispostion in the Defense Waste Processing Facility.

A.3.16 Y-12 Plant (Y-12)

The materials listed in Table A.21 [based on Keck (Sept. 11, 1996)] are stored at the Oak Ridge Y-12 Plant. Because there are currently no planned requirements or needs for these materials, they are considered surplus. The Y-12 Plant ^{233}U materials are considered to be in a classified configuration. To be transferred to ORNL (Building 3019) for storage, these materials either would need to be cast in an unclassified shape or have the information about their present shape declassified.

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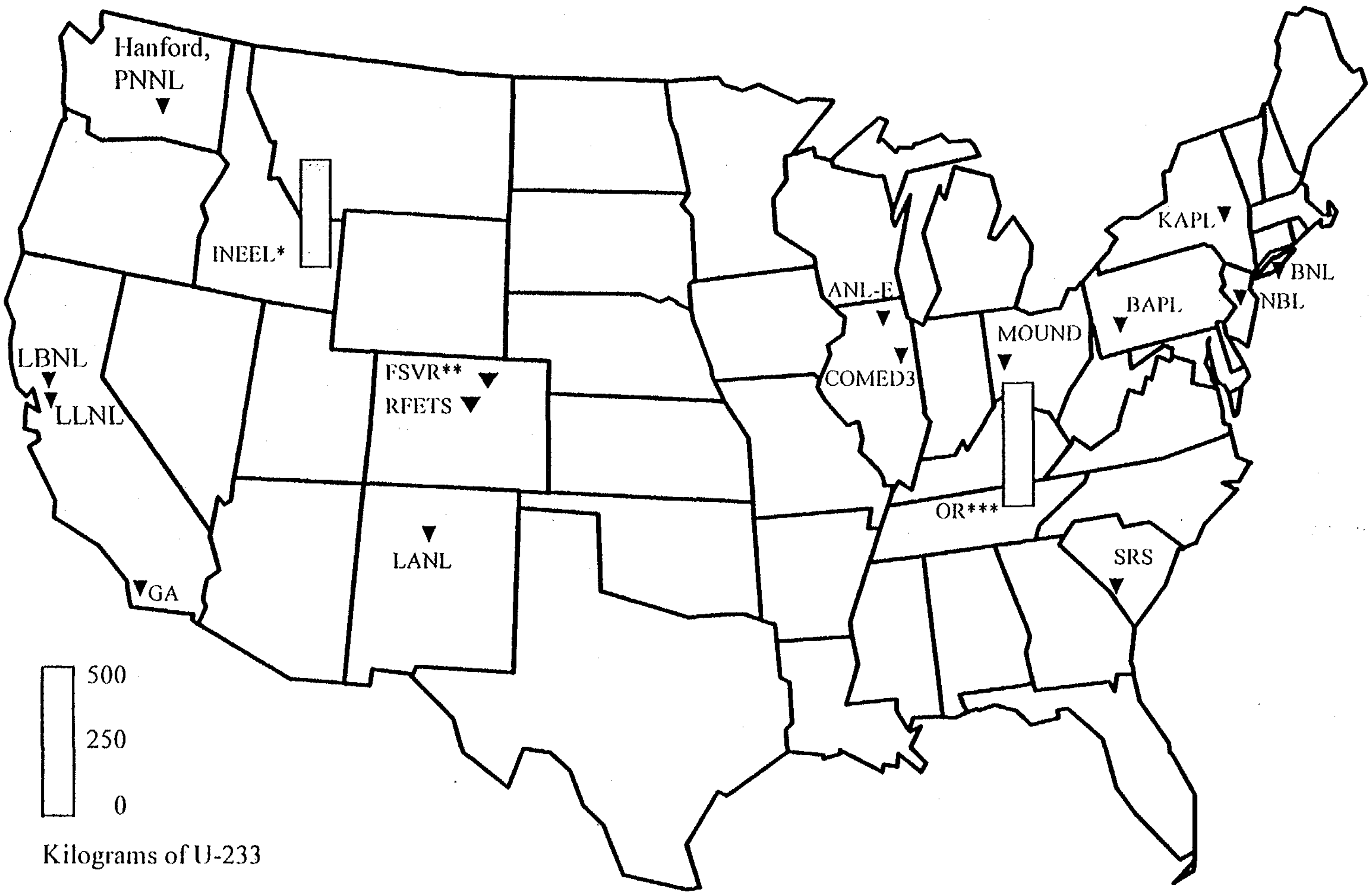

*Excludes contributions from irradiated SNF owned by DOE. Includes very small contributions from ANL-W.
**FSVR irradiated SNF owned by DOE.
***Includes contributions from ORNL and Y-12.
Note. Only locations are shown for sites whose U-233 inventories are either classified or are less than 10 kg.

Fig. A.1. Major locations and masses of current U-233 inventories.

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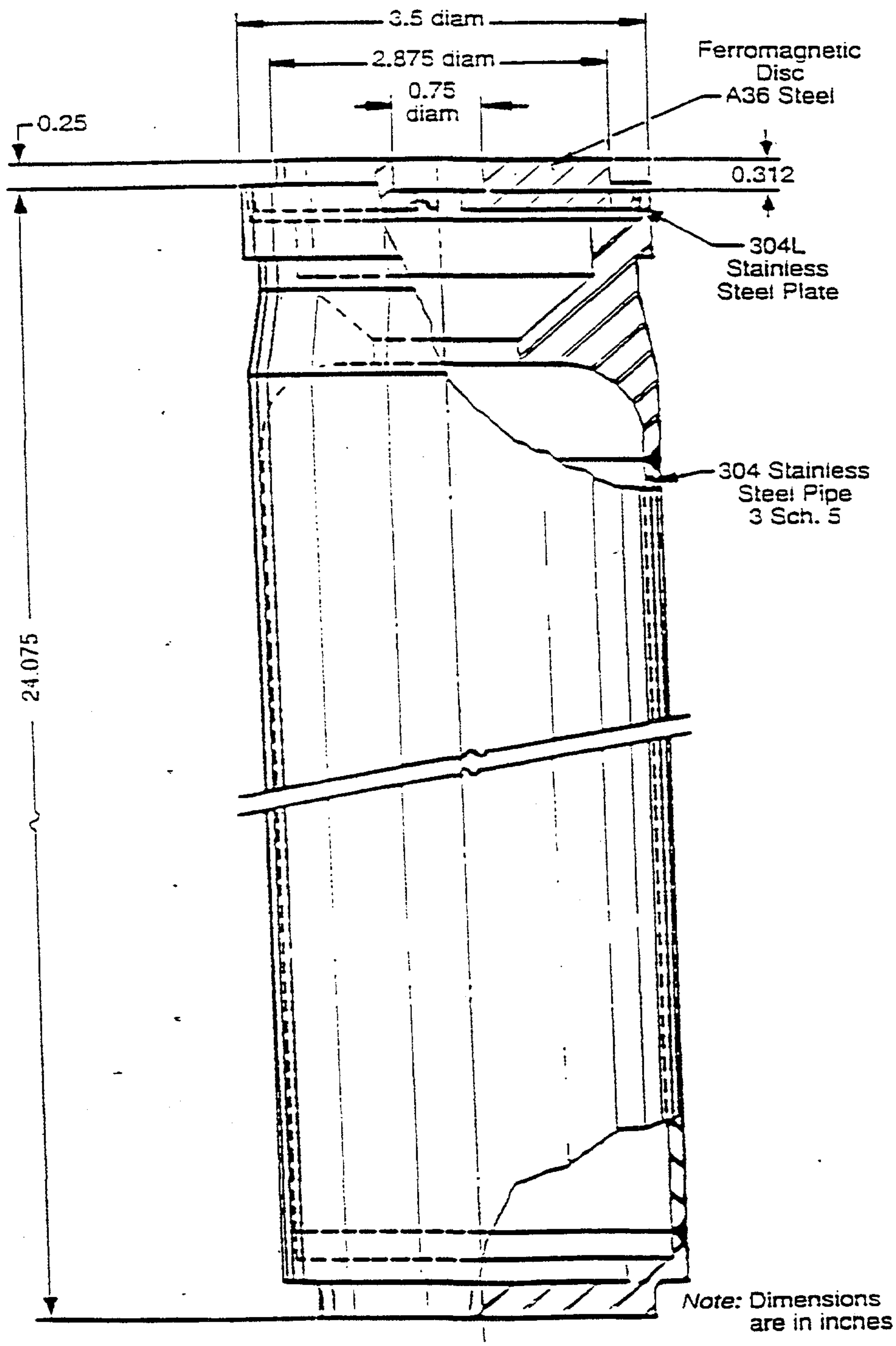

Fig. A.2. Storage-can assembly for CEUSP solidified waste.

Table A.1. Domestic sites that have accountable quantities of ^{233}U materials

Site	Acronym
DOE sites	
Argonne National Laboratory–East	ANL–E
Argonne National Laboratory–West	ANL–W
Bettis Atomic Power Laboratory	BAPL
Brookhaven National Laboratory	BNL
Hanford Site	Hanford
Idaho National Engineering and Environmental Laboratory–	
Idaho Chemical Processing Plant	INEEL–ICPP
Radioactive Waste Management Complex	INEEL-RWMC
Knolls Atomic Power Laboratory	KAPL
Lawrence Berkeley National Laboratory	LBNL
Lawrence Livermore National Laboratory	LLNL
Los Alamos National Laboratory	LANL
New Brunswick Laboratory	NBL
Mound Plant	Mound
Oak Ridge National Laboratory	ORNL
Pacific Northwest National Laboratory	PNNL
Rocky Flats Environmental Technology Site	RFETS
Savannah River Site	SRS
Y-12 Plant	Y-12
NRC-licensed sites	
Commonwealth Edison: Dresden Reactor–Unit 3	COMED3
Fort St. Vrain Reactor	FSVR
General Atomics Laboratory, San Diego	GA

Table A.2.. Summary of domestic ^{233}U material characteristics and inventories[a]

Site	Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
				^{233}U	^{235}U			
ANL–E	MP, MI, SN, RS, RU	B1, B2, C2, G3, P0, V5, X2	5	71, 72, 74		0.028	0.028	0
ANL–W	MA, MP, PI, PO, SN, SS, SZ	B1, C0, C2, D1, G0, G1 P0, P1, P3, V1, V5, V7	63	71, 72		0.155	0.154	0
BAPL	RO, HO, SR	C4, D1, *f*	13	71, 72, 74	38	0.427	0.405	0.014
COMED3			0			0	0	0
FSVR[g]			0			0	0	0
GA	SS, SO	C2, C4	2	72	38	0.00041	0.00021	0.00019
Hanford	PO, PI, SN	B1, C0, W1	3	71		0.597	0.079	0
INEEL/ICPP	PO, RO, RU, SN	X2	213	72, 73		358.6	351.6	0
LANL	PO, PI, MP, MI, MA, UO, SO, SS, CP, NM, RO, SN, OO	B1, B2, C0, C1, C2, C3, C4, F2, G1, P0, P1, U0, U1, U2, V1, V5, V7, W4, and X2	109	72		7.243	7.105	0
LLNL	MP, MA, PO, PI, RO	C0, C2	50	72		3.321	3.253	0
Mound[h]			0			0	0	0
ORNL	MA, UO, OO, MP, PO	C3, C4, V1, X1	1,054	71–74	36	1,387.709	427.341	796.334
PNNL	MA, PO, PI, SN, SZ	C0, C1, C2, G2, P0	15	71, 72		0.048	0.047	0
RFETS	CP, NO	D1	5	72		0.004	0.004	0
SRS			0			0	0	0

Table A.2.. Summary of domestic ^{233}U material characteristics and inventories[a]

Site	Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
				^{233}U	^{235}U			
Y-12	MP, PI	X1	5	*i*	37	42.6	0.8	38.7
Total			1,537			1,800.7	790.8	835.0

[a]Excludes contributions from irradiated SNF.
[b]Material-type and form codes listed in Table A.3 of Appendix A.
[c]Packaging types and codes listed in Table A.4 of Appendix A.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5 of Appendix A.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5 of Appendix A.
[f]Entrained in equipment.
[g]Excludes contributions from irradiated SNF owned by DOE.
[h]All Mound ^{233}U material was shipped to ORNL in 1996. This material is now included in the ORNL inventory.
[i]Information not available.

Table A.3. Material-form code definitions

Material type	Form	Code
Weapon component	Parts	PA
	Pits	PT
	Canned subassembly (CSA)[a]	CS
Metal	Pure	MP
	Impure	MI
	Alloys	MA
Oxides	Pure	PO
	Impure	PI
	Other (specify)	OO
Compounds	Uranium hexafluoride	UF
	Other (specify)	UO
Sources and samples	Sealed	SS
	Other (specify)	SO
Combustibles	Graphite	CG
	Paper, plastics, wood, mop heads, etc.	CP
	Other (specify)	CO
Noncombustibles	Glass	NG
	Nonuranium metal	NM
	Other (specify)	NO
Process residues	Reduction	RR
	Incinerator ash	IA
	Sludge	SR
	Filters	RF
	Other (specify)	RO
Solutions	Nitric acid	SN
	Basic	SC
	Organic	OS
	Other (specify)	SZ
Reactor fuel	Unirradiated	RU
	Targets	RT
	Slightly irradiated	RS
	Other (specify)	RO
Hold-up	Materials in pipes, tanks, ducts, equipment, etc.	HO

[a]When seal is broken, canned subassembly is called *parts*.

Table A.4. Packaging types and codes

General description	Packaging code	Subcode	Packaging details
Cans	C	0	Food-pack/rim seal (tinned)
		1	Food-pack/rim seal (stainless)
		2	Slip lid
		3	Screw lid
		4	Other (specify)
Plastic bagging	B	0	Unknown
		1	Polyethylene
		2	Polyvinylchloride (PVC)
		3	Other (specify)
Metal foil	F	0	Unknown
		1	Aluminum
		2	Lead
		3	Other (specify)
Vessels	V	0	Unknown
		1	Welded
		2	Knife-edge seal (i.e., Conflat®)
		3	Elastomeric seal (O-ring)
		4	Compression seal (Swagelock® etc.)
		5	Screw lid
		6	Gas cylinder (UF_6)
		7	Other (specify)
Glass	G	0	Other (specify)
		1	Screw lid
		2	Sealed vials/capsules
		3	Glass-metal seal
Plastic containers	P[a]	0	Polyethylene/polypropylene–sealed
		1	Polyethylene/polypropylene–unsealed
		2	Polyethylene/polypropylene–unknown
		3	Other (specify)
Unknown	U	0	Unknown
		1	Suspected to be metal
		2	Suspected to be plastic
		3	Other (specify)
Drums	D	1	55-gal
		2	30-gal
		3	<30-gal
		4	Unspecified (add V if vented)
Tanks	T	0	Unknown
		1	Raschig ring–filled
		2	Geometrically favorable
		3	Other (specify)

Table A.4 (continued)

General description	Packaging code	Subcode	Packaging details
Wooden crates or boxes	W	0	Metal burial box
		1	Cardboard
		2	Wooden
		3	Fiberglass
		4	Other (specify)
Shipping containers and overpacks	X	0	5A overpack
		1	6M 110-gal
		2	Other (specify)
		3	Birdcage (storage only)

[a]P = bottle.

Table A.5. Nuclear material-type codes

Type code	Type description	Reporting unit	Type code	Type description	Reporting unit
	Uranium—depleted in ^{235}U, wt %		44	^{241}Am	g
10	Total	kg	45	^{243}Am	g
11	<0.21	kg	46	Curium	g
12	0.21–<0.24	kg	47	Berkelium	µg
13	0.24–<0.26	kg	48	Californium	µg
14	0.26–<0.28	kg		Plutonium	
15	0.28–<0.31	kg	50	Total	g
16	0.31–<0.50	kg		^{240}Pu	
17	0.50–<0.60	kg	51	<4.00	g
18	0.60–<0.711	kg	52	4.00–<7.00	g
	Uranium—enriched in ^{235}U, wt %		53	7.00–<10.00	g
20	Total	g	54	10.00–<13.00	g
21	>0.711–<0.90	g	55	13.00–<16.00	g
22	0.90–<1.15	g	56	16.00–<19.00	g
23	1.15–<1.60	g	57	19.00 and above	g
24	1.60–<2.00	g		*Lithium—enriched in ^{6}Li*	kg
25	2.00–<2.60	g	60	Total	kg
26	2.60–<2.90	g	61	>Normal to <55.00	kg
27	2.90–<3.10	g	62	55.00–<80.00	kg
28	3.10–<3.40	g	63	80.00 and above	kg
29	3.40–<3.90	g		*Uranium—enriched in ^{233}U*	
30	3.90–<4.10	g	70	Total	g
31	4.10–<5.00	g	71	<5 ppm ^{232}U	g
32	5.00–<10.00	g	72	5–<10 ppm ^{232}U	g
33	10.00–<20.00	g	73	10–<50 ppm ^{232}U	g
34	20.00–<35.00	g	74	50 ppm and above ^{232}U	g
35	35.00–<45.00	g	81	Normal uranium (0.711 wt % ^{235}U)	kg
36	45.00–<80.00	g	82	^{237}Np	g
37	80.00–<92.00	g	83	^{238}Pu	g (1×10^{-1})
38	92.00–<94.00	g	86	D_2	kg (1×10^{-1})
39	94.00 and above	g	87	Tritium	g (1×10^{-2})
	^{242}Pu		88	Thorium	kg
40	Total	g	89	Uranium in cascades	g
41	20–60	g	90	This series available for local use	
42	>60	g			

Table A.6. Summary of ^{233}U material characteristics at ANL–E[a]

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e,f] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
Foil discs (MP)	G3, B1, C2	1	72	*g*	*	*	0
Discs (MI)	G3, C2	1	71	*g*	*	*	0
Solutions (SN)[h]	P0, V5	1	71	*g*	*	*	0
Reactor fuel (RS)[i]	G2, X2[j]	1	72	*g*	*	*	0
Reactor fuel (RU)	C2, B2, P0	1	74	*g*	*	*	0
Total	*k*	5	71, 72, 74	*g*	0.028	0.028	0

[a]Based on Lambert (July 11, 1996) and Barkalow and Poupa (Aug. 12, 1996).
[b]Material-type and form codes listed in Table A.3. Excludes 8 g of reactor-irradiated nuclear material.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]An asterisk (*) is used to represent mass quantities of material less than 0.1 kg. The total of listed ^{233}U is 0.028 kg.
[g]The ^{233}U material is enriched to greater than 98%. There are no accountable quantities of ^{235}U material involved.
[h]Mixed solution of ^{233}U and nitric acid (HNO_3).
[i]Five fuel pins from Pacific Northwest National Laboratory.
[j]The material, slightly irradiated reactor fuel, is contained in a quartz vial that has been positioned in a lead pot which, in turn, has been placed in a DOT Specification 2R vessel (described in 49 CFR 178.360) for storage.
[k]B1, B2, C2, G3, P0, V5, and X2.

Table A.7. Summary of ^{233}U material characteristics at ANL–W[a]

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e,f] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
Metal (MA)	P0	1	72		*	*	0
Metal (MA)	P1	3	72		*	*	0
Metal (MP)	P1	21	72		*	*	0
Metal (MP)	P3	1	72		*	*	0
Metal (MP)	C0, V1	1	71		0.1	0.1	0
Metal (MP)	C2, V1	1	72		*	*	0
Metal (MP)	P3, V1	1	72		*	*	0
Oxides (PI)	B1	1	72		*	*	0
Oxides (PI)	C2	12	72		*	*	0
Oxides (PI)	G0, V7	2	71		*	*	0
Oxides (PO)	C2, V5	1	72		*	*	0
Solutions (SN)	G0 (flask)	2	72		*	*	0
Solutions (SN)	G0 (tubes)	5	72		*	*	0
Solutions (SN)	G1	2	71		*	*	0
Solutions (SN)	P0	1	72		*	*	0
Solutions (SN)	P0	2	72		*	*	0
Solutions (SN)	P0	4	72		*	*	0
Solutions (SZ)	P0, D1	1	72		*	*	0
Sources (SS)	P0	1	72		*	*	0
Total	g	63	71, 72		0.155	0.154	0

[a]Based on Meppen (Aug. 21, 1996) and Haga (Jan. 23, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]An asterisk (*) is used to represent mass quantities of material less than 0.1 kg.
[g]B1, C0, C2, D1, G0, G1, P0, P1, P3, V1, V5, and V7.

Table A.8. Summary of ^{233}U material characteristics at BAPL[a]							
Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
RO[f]	D1	1	71, 72		0.01232	0.01218	0
RO[g]	D1	3	72		0.02253	0.02215	0
HO	*h*	2	72		0.37300	0.36655	0
SR	C4[i,j]	1	71	38	0.00504	0.00027	0.00446
SR	C4[i,j]	1	72, 74	38	0.00455	0.00336	0.00109
SR	C4[i,j]	3	72		0.00058	0.00058	0
SR	C4[i,j]	2	72	38	0.00894	0.00028	0.00812
Total	*k*	13	71, 72, 74	38	0.42696	0.40537	0.01367

[a]Based on Johnson (Sept. 17, 1996) as amended by Stevenson (June 12,1998). Includes TRU waste materials which are stored in three 17C 55-gal drums, one 17C 5-gal. pail, and entrained in two equipment grinders. The mass of all of these waste materials is currently known to be in excess of 5603 kg.

[b]Material-type and form codes listed in Table A.3.

[c]Packaging types and codes listed in Table A.4.

[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.

[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.

[f]Chemical waste (solid) samples.

[g]Empty containers, rags, plastic (poly), and noncompressible waste.

[h]Entrained in equipment (i.e., two grinders).

[i]Paint cans.

[j]Irradiated material.

[k]C4, D1, and paint cans.

Table A.9. Summary of ^{233}U material characteristics at COMED3[a]

(Note: This is all SNF.)

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
RO[f]	C4[g]	1	70	20	0.5	0.3	0.2
Total[h]		0			0	0	0

[a]Based on Larson (Sept. 18, 1996).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]Eighteen irradiated fuel rods.
[g]Removable lid.
[h]Because the material listed is irradiated SNF, it is not included in the DOE complex total for ^{233}U candidate disposition materials (see Table A.2).

Table A.10. Summary of ^{233}U material characteristics at FSVR[a]

(Note: This is all SNF.)

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e,f] (kg)	^{235}U[e,f] (kg)
			^{233}U	^{235}U			
Reactor fuel (RO)[g]	Other (X2)[h]	244	70, 71, 72, 73, 74	33, 34, 35, 36, 37	822.4	236.0	404.5
Reactor fuel (RO)[i]	Other (X2)[h]	2	70[j]	E3[j]	*	*	*
Total[k]		0			0	0	0

[a]Based on Borst (July 2, 1996).

[b]Material-type and form codes listed in Table A.3. Consists of irradiated fuel.

[c]Packaging types and codes listed in Table A.4.

[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.

[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.

[f]An asterisk (*) is used to represent mass quantities of material less than 0.1 kg.

[g]Highly irradiated Fort St. Vrain (FSV) high-temperature, gas-cooled reactor (HTGR) fuel elements or fuel handing units (FHUs). The HTGR fuel consists of both thorium carbide particles and uranium and thorium carbide particles encased in a modified three-layer (actually four-layer) coating [called a tristructural isotropic (TRISO) coating], which is bonded into fuel rods and inserted into hexagonal graphite blocks.

[h]A Fuel Storage Container (FSC), which is a custom-designed carbon-steel tube, closed at the lower end and sealed at the top. Each FSC is about 18 in. in diam and about 190 in. long, holds six FHUs, and is shipped in a Nuclear Regulatory Commission (NRC)-assigned Transnuclear (TN-FSV) cask.

[i]This fuel is foreign-owned and contained in two FHUs. The HTGR fuel elements have a mixed ownership between the DOE and the French Atomic Energy Commission (Commissariat a l'Energie Atomique). The U.S.-owned fuel is inseparable from the foreign-owned fuel.

[j]The codes used for these entries pertain to privately owned material. For ^{235}U, E3 denotes French-owned ^{235}U that has a ^{235}U assay of 20% or more, but less than 80%. For this material, estimated equivalent DOE codes would be 74 for ^{233}U and 33 for ^{235}U.

[k]Because the material listed is irradiated SNF, it is not included in the DOE complex total for ^{233}U candidate disposition materials (see Table A.2).

Table A.11. Summary of ^{233}U material characteristics at GA[a]

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e,f] (kg)	^{235}U[e,f] (kg)
			^{233}U	^{235}U			
Sources (SS)[g]	Cans (C4)	1	72	38	0.00030	0.00010	0.00019
Sources (SO)[h]	Cans (C2)	1	72		0.00011	0.00011	0
Total (SS, SO)	Cans (C2, C4)	2	72	38	0.00041	0.00021	0.00019

[a]Based on Wisham (Aug. 21, 1996, and May 12, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]An asterisk (*) is used to represent mass quantities of material less than 0.1 kg.
[g]Dosimeter (made of stainless steel, 0.5 in. diam × 4 ft long) containing metal wires and coated particles with double-encapsulated microspheres of ^{239}Pu, ^{235}U, ^{238}U, and ^{232}Th containing oxide and carbides.
[h]Coated particles.

Table A.12. Summary of ^{233}U material characteristics at Hanford[a]

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
Oxides (PO)	C0	1	71		0.037	0.037	0
Oxides (PI)	B1, W1	1	71		0.521	0.039	0
Solutions (SN)	B1, W1	1	71		0.039	0.003	0
Total	B1, C0, W1	3	71		0.597	0.079	0

[a]Based on Serier (Aug. 21, 1996) and Hulse (Mar. 25, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material type codes listed in Table A.5.

Table A.13.a Summary of unirradiated ^{233}U material characteristics at INEEL–ICPP[a]							
Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e,f] (kg)	^{235}U[e,f] (kg)
			^{233}U	^{235}U			
RO[g]	C4	6	71		*	*	0
RO[g]	C4	14	72		*	*	0
RO[g]	C4	1	73		*	*	0
PO[h]	C3	1	71		*	*	0
PO[h]	C3	1	71		*	*	0
SN[h]	P3	3	71		*	*	0
PO[h]	C4	1	71		*	*	0
SN[h]	P3	1	71		*	*	0
RU[h]	C0	6	71		*	*	0
PO[h]	B1	39	71	39	*	*	*
PO[h]	C0	1	72		*	*	0
RU[i]	X2	1	72		16.84	16.56	0
RU[i]	X2	40	72		306.64	300.80	0
RU[i,j]	D1,X1	107	73		18.2	17.7	0
RU[i,j]	D1	12	73		1.7	1.682	0
RU[i,j]	X1	53	73		15.52	14.818	0
Total[k]	X2	287	71, 72, 73		358.6	351.6	0

[a]Based on Chemical Technology Division (Sept. 22, 1995); DOE Idaho Operations Office (July 19, 1995); Laible (Aug. 1, 1996); and Bright (Nov. 12, 1996).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See Table A.5.
[f]An asterisk (*) is used to represent mass quantities of material less than 0.1 kg.
[g]Scrap from Bettis Atomic Power Laboratory
[h]Miscellaneous lab materials.
[i]Unirradiated Shippingport light-water breeder reactor (LWBR) fuel, the composition of which is described in Appendix B.
[j]Located at the INEEL Radioactive Waste Management Complex (RWMC).

Table A.13.b. Summary of irradiated ^{233}U material characteristics at INEEL–ICPP[a]							
Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
RO[f]	X2	47	74	34, 36, 37, 38	550.4	508.6	10.1
RO[g]	X2	38	71		10.5	0.8	0
RO[g]	X2	634	72		168.4	15.3	0
RO[g]	X2	193	73		36.5	5.8	0
RO[g]	X2	737	74		116.6	24.4	0
RO[h]	X2	84	71		31.1	3.7	0
RO[h]	X2	78	72		30.0	5.2	0
RO[h]	X2	376	73		154.8	45.0	0
RO[h]	X2	206	74		92.3	36.3	0
Total	X2	2393	71, 72, 73, 74		1190.6	608.8	10.1

[a]Based on Chemical Technology Division (Sept. 22, 1995); DOE Idaho Operations Office (July 19, 1995); Laible (Aug. 1, 1996); and Bright (Nov. 12, 1996).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See Table A.5.
[f]Irradiated LWBR SNF.
[g]Peach Bottom fuel (Core 1 and 2).
[h]Fort St. Vrain Reactor (FSVR) fuel.

Table A.14. Summary of ^{233}U material characteristics at LLNL[a]							
Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
MP	C0, C2[f]	9	72		0.462	0.454	0
MA	C0, C2	5	72		0.034	0.033	0
PO	C0, C2	21	72		2.570	2.516	0
PI	C0, C2	3	72		0.230	0.225	0
RO	C0, C2	12	72		0.025	0.025	0
Total	C0, C2	50	72		3.321	3.253	0

[a]Based on Maslin (Aug. 29, 1996).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material type codes listed in Table A.5.
[f]Paint cans.

Table A.15. Summary of ^{233}U material characteristics at LANL[a]							
Material form description[b]	Packaging types[c]	Number of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
UO[f]	U0, C0	3	72		0.049	0.048	0
PO	G1, P1, C2; C2, B2, C2	3	72		0.522	0.509	0
SO	P1, C1	1	72		0.001	0.001	0
SS	U2, C2	2	72		0.234	0.230	0
PI	C4; P0, B1; V1, C2, C2; U0, C0; V1, C2	11	72		1.604	1.573	0
PI[f]	U0	6	72		0.461	0.449	0
UO[g]	U0	2	72		0.271	0.265	0
UO[h]	C2, B2, C2	1	72		0.001	0.001	0
PI[i]	C2, V7; P0, B1	3	72		0.240	0.236	0
OO[j]	C2, V1	3	72		0.005	0.005	0
PI[j]	C4	1	72		0.700	0.686	0
CP[k]	C2, B2, C2	1	72		0.006	0.006	0
SN[l]	G1	1	72		0.001	0.001	0
MP	V5; W4; U2, C2; U0, C0	21	72		0.171	0.171	0
MP[m]	U2, C2	1	72		0.003	0.003	0
MI	V1, C2; C2, B2, C2; C2, B2, V1; V1, C2; U0, C0; C2; U0, C2; U1, C2; U2, X2	27	72		2.309	2.270	0
MI[f]	C0, U0	3	72		0.017	0.017	0
MI[n]	C0, U0	1	72		0.005	0.005	0
MA	C2, B2, C2	1	72		0.188	0.183	0
MA[m]	C2, B2, V1	1	72		0.062	0.061	0
MA[o]	C2, C3, F2	7	72		0.068	0.067	0
NM[p]	U0	1	72		0.118	0.115	0
NM	G1, P1, C2	1	72		0.011	0.011	0
RO[q]	C2, B2, C2	3	72		0.187	0.183	0
RO[r]	G1, P1, C2; P0; U0, C0	4	72		0.009	0.009	0
Total	s	109	72		7.243	7.105	0

Table A.15. Summary of ^{233}U material characteristics at LANL[a]

Material form description[b]	Packaging types[c]	Number of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			

[a]Based on Martinez (Mar. 4, 1997, and July 7, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]Plutonium contaminated.
[g]Impure nitrate.
[h]Impure tetrafluoride.
[i]Trioxide.
[j]U_3O_8.
[k]Cellulose rags.
[l]Nitrate standard.
[m]Encapsulated.
[n]Metal turnings.
[o]History samples.
[p]Graphite.
[q]Hydroxide precipitate.
[r]Process residue sweepings.
[s]B1, B2, C0, C1, C2, C3, C4, F2, G1, P0, P1, U0, U1, U2, V1, V5, V7, W4, and X2.

Table A.16. Summary of ^{233}U material characteristics at Mound[a]

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
PO	X1	20	70		3.5	3.5	0.0
Total[f]		0			0.0	0.0	0.0

[a]Based on Stadler (Sept. 16, 1996).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material type codes listed in Table A.5.
[f]All of the ^{233}U material at Mound was shipped to ORNL in 1996. This material is now included in the ORNL inventory.

Table A.17. Summary of ^{233}U material characteristics at ORNL[a]							
Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
Uranium metal and alloys (MA)	C4[f]	29	71–73		17.310	16.955	0
Salts (UO)	C3	4	74		3.191	2.919	0
UO_x powder (OO)	C4[f]	44	71–73		10.382	10.166	0
	C	22	71		15.358	15.015	0
	V1[g]	128	72		45.674	44.766	0
	C3	142	72		96.453	91.153	0
	C4[h]	27	73		11.143	10.720	0
	C4[h]	140	74		67.371	61.569	0
U_3O_8 monolith (OO)	C4[i]	27	73		65.188	60.265	0
	C4[i]	403	74	36	1,042.585	101.143	796.334
Mound (PO)[j]	X1	20	71–73		3.648	3.493	0
Miscellaneous (MP)[k]	C4[f]	68	71, 72		9.217	9.018	0
Total	C3, C4, V1, X1	1054	71–74	36	1387.520	427.182	796.334

[a]Based on Krichinsky (Aug. 25, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]Cans of various types.
[g]Stainless steel plates welded shut.
[h]Welded aluminum cans.
[i]Stainless steel cans welded shut.
[j]Material (UO_x powder) shipped from the Mound Plant in 1996.
[k]Uranium in irregular forms.

Table A.18. Summary of ^{233}U material characteristics at PNNL[a]							
Materia l form descript ion[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e,f] (kg)	^{235}U[e,f] (kg)
			^{233}U	^{235}U			
SN	G2	7	71		*	*	0
PO	C1	1	71		*	*	0
SZ	C2	1	71		*	*	0
SN[g]	C2	1	71		*	*	0
PO[g]	C2	1	72		*	*	0
PO[g]	C2	1	71		*	*	0
PI	C0	1	71		*	*	0
MA	C2	1	72		*	*	0
SN[g]	P0	1	71		*	*	0
Total	MA, P0, PI, SN, SZ	15	71, 72		0.048	0.047	0

[a]Based on Andre (May 2, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]An asterisk (*) is used to represent mass quantities less than 0.1 kg.
[g]Material that is in programmatic use (not excess).

Table A.19. Summary of ^{233}U material characteristics at RFETS[a]

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U[f] (kg)	^{233}U[e,f] (kg)	^{235}U[e,f] (kg)
			^{233}U	^{235}U			
CP	D1	3	72		*	*	0
NO[g]	D1	2	72		*	*	0
Total	D1	5	72		0.004	0.004	0

[a]Based on James (Mar. 10, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]An asterisk (*) is used to represent mass quantities less than 0.1 kg.
[g]Consists of light (nonfissile) metals.

Table A.20. Summary of ^{233}U material characteristics at SRS[a]

(Note: All SRS ^{233}U inventory is contained in irradiated SNF, including 17 thorium target slugs.)

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
		0			0	0	0
Total		0			0	0	0

[a]Based on Severynse (Sept. 4, 1996).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.

Table A.21. Summary of ^{233}U material characteristics at Y-12[a]

Material form description[b]	Packaging types[c]	No. of packages	Material-type code[d,e]		Total U (kg)	^{233}U[e] (kg)	^{235}U[e] (kg)
			^{233}U	^{235}U			
MP	X1	3	72[f]	37	30.9	0.6	28.1
PI	X1	2	72[f]	37	11.7	0.2	10.6
Total	X1	5	72[f]	37	42.6	0.8	38.7

[a]Based on Keck (Sept. 11, 1996) and Hall (Apr. 1, 1997).
[b]Material-type and form codes listed in Table A.3.
[c]Packaging types and codes listed in Table A.4.
[d]Nuclear material-type codes listed for ^{233}U and ^{235}U in Table A.5.
[e]Accountable amounts only. See nuclear material-type codes listed in Table A.5.
[f]Six parts per million of ^{233}U is ^{232}U.

Table A.22. Characteristics of unirradiated Shippingport LWBR fuel at INEEL–ICPP[a,b]

Description
Unirradiated fuel rods from the Shippingport light-water breeder reactor

Storage location
INEEL–ICPP Dry Wells Area (Facility 749: underground dry storage facilities for fuel)

Composition
UO_2 ceramic fuel pellets with Th, Zr, and Ca oxides; Zr-clad and Th blanket

Material inventory (kg)

Total uranium	323.5
^{235}U (0 wt %)	0.0
^{233}U (98 wt %)	317.4
^{232}U (5–10 ppm)	<<0.1
Total thorium (Th)	13,679.0

Storage containers
40 stainless steel canisters containing a total of 14,468 unirradiated fuel rods, some of which are illustrated in Fig. A.2. The canisters are placed in underground dry storage vaults (also called silos), with each vault typically containing 2 canisters

Disposition
This material was originally brought to INEEL for storage. Present plans call for continued storage and monitoring of the material until it can be shipped to a SNF or HLW repository. There are no plans to use this fuel[a]

[a]Based on U.S. DOE Idaho Operations Office (July 19, 1995) and Detrick (May 6, 1997 and April 8, 1998).
[b]Excludes unirradiated fuel rods stored at the INEEL–RWMC.

Table A.23. Characteristics of Consolidated Edison Uranium Solidification Program (CEUSP) material at ORNL[a]

Description	
Monolithic uranium oxide material (radioactive and hazardous)	
Storage location	
Radiochemical Processing Plant (Building 3019, ORNL)	
Material inventory, kg	
Total material	1673.3
Total U	1042.6
^{235}U	796.4
^{233}U	101.1
Uranium isotopic composition, wt %	
^{233}U	9.69
^{234}U	1.39
^{235}U	76.52
^{236}U	5.60
^{238}U	6.80
^{232}U	~0.01 (about 110 ppm)
Chemical composition, wt %	
U_3O_8	64.1
CdO[b]	19.6
Gd_2O_3[b]	2.2
Metal contaminants	14.1 (includes Si, Fe, Al, P, and Cr)
Storage containers	
403 welded canisters; each inner can is placed inside a tin-plate outer can (see Fig. A.2)	
Inner can: 3.5 in. OD by 24.25 in. length	
Outer can: 3.625 in. OD by 24.75 in. length	
Average radiation levels from storage containers	
At surface: 300–350 rem/h	
At 1 ft from surface: 60–80 rem/h	

[a]Based on Peer Consultants, P.C., and Engineering, Design, and Geosciences Group, Inc. (Dec. 23, 1987).
[b]Neutron poisons cadmium and gadolinium were added to the CEUSP material to reduce the risk of a criticality accident during its 17-year period of storage as a liquid.

Table A.24. List of site ^{233}U information and material management sources

Site	Person(s) responsible for information/data	Person(s) responsible for material management
	DOE sites	
ANL–E	Thomas Barkalow University of Chicago Environment, Safety, and Health Division 9700 South Cass Avenue Building 201 Argonne, IL 60439 Phone: 630/252-9243 Fax: 630/252-5778	Kenneth W. Poupa
	Kenneth W. Poupa University of Chicago Special Materials Group 9700 South Cass Avenue Building 201, Mail Stop OCF-SPM Argonne, IL 60439 Phone: 630/252-7776 Fax: 630/252-6397	
ANL–W	Roger D. Haga R. L. Powell University of Chicago Safeguards Section P.O. Box 2528 Building 752, Mail Stop 6000 Idaho Falls, ID 83403-2528 Phone: 208/533-7070 Fax: 208/533-7599	Roger D. Haga
BAPL	Earl G. Johnson Renee A. Stevenson Westinghouse Electric Corporation Laboratory Operational Safeguards Zap 62N/LOS P.O. Box 79 West Mifflin, PA 15122-0079 Phone: 412/476-7347 Fax: 412/476-5687	David K. Dedrick Kathy K. Mihalco Westinghouse Electric Corporation Development Laboratories Operations Zap 64C/DLO P.O. Box 79 West Mifflin, PA 15122-0079 Phone: 412/476-7453 Fax: 412/476-7512
Hanford	Marjory N. Serier Westinghouse Hanford Company Safeguards and Security Mail Stop G3-40 P.O. Box 1970 Richland, WA 99352 Phone: 509/372-1444 Fax: 509/376-4312	Alan A. Grasher Babcock & Wilcox Hanford Company Transition Projects Mail Stop S6-15 P.O. Box 1970 Richland, WA 99352 Phone: 509/373-2565 Fax: 509/372-3402

Table A.24 (continued)

Site	Person(s) responsible for information/data	Person(s) responsible for material management
Hanford (contd.)	Gary B. Hulse Babcock & Wilcox Hanford Company Mail Stop A4-25 P.O. Box 1970 Richland, WA 99352 Phone: 509/372-1481 Fax: 509/376-0887	
INEEL/ICPP	Ernie L. Laible Lockheed Martin Idaho Technologies Co. Safeguards/Material Control and Accountability Mail Stop 5102 P.O. Box 1625 Idaho Falls, ID 83415-5102 Phone: 208/526-2113 Fax: 208/526-5432	Mary Alice Thom Lockheed Martin Idaho Technologies Co. Safeguards/Materials Management Willow Creek Building Mail Stop 3226 P.O. Box 1625 Idaho Falls, ID 83415-3226 Phone: 208/526-2737 Fax: 208/526-1444
LLNL	Brent H. Ives University of California Materials Management Division P.O. Box 808 Mail Stop L-347 Livermore, CA 94550 Phone: 925/423-2636 Fax: 925/423-1685	William R. Ruvalcaba University of California Materials Management Division P.O. Box 808 Mail Stop L-347 Livermore, CA 94550 Phone: 925/423-4326 Fax: 925/423-1685
	Karen C. Maslin University of California Materials Management Division P.O. Box 808 Mail Stop L-347 Livermore, CA 94550 Phone: 925/422-0463 Fax: 925/423-1685	
LANL	Audrey L. R. Martinez University of California Nuclear Materials and Stockpile Management Division Nuclear Materials Management Office P.O. Box 1663 Mail Stop E524 Los Alamos, NM 87545 Phone: 505/665-7333 Fax: 505/665-8997	Randall M. Erickson Program Manager Nuclear Materials Stockpile Management Stabilization and Disposition P.O. Box 1663 Mail Stop F660 Los Alamos, NM 87545 Phone: 505/667-4950 Fax: 505/667-1777

Table A.24 (continued)

Site	Person(s) responsible for information/data	Person(s) responsible for material management
Mound	Thomas A. Grice EG&G Mound Applied Technologies, Inc. Safeguards & Security Administration Material Control & Accountability Group Mail Stop T-248 P.O. Box 3000 Miamisburg, OH 45343-3000 Phone: 513/865-3371 Fax: 513/847-5264	Ronald G. Bonifay EG&G Mound Applied Technologies, Inc. Safeguards & Security Administration Material Control & Accountability Group Mail Stop T-248 P.O. Box 3000 Miamisburg, OH 45343-3000 Phone: 513/865-3537 Fax: 513/847-5264
ORNL	Alan M. Krichinsky Lockheed Martin Energy Research Corp. Chemical Technology Division Radiochemical Development Facility P.O. Box 2008 Building 3019 Mail Stop 6046 Oak Ridge, TN 37831-6046 Phone: 423/574-6940 Fax: 423/576-8284	Alan M. Krichinsky
PNNL	James P. Andre Scott W. Gority Safeguards and Security Services Battelle Memorial Institute Pacific Northwest National Laboratory Mail Stop K6-44 P.O. Box 999 Richland, WA 99352 Phone: 509/376-0251 or 9985 Fax: 509/376-6802	James P. Andre Scott W. Gority
RFETS	Mary P. Rodriguez Charles A. Finleon Rocky Flats Environmental Technology Site Safeguards and Accountability Building 750 P.O. Box 464 Golden, CO 80402-0464 Phone: 303/966-2052 or 2066 Fax: 303/966-5683	Scott Davies William H. James Safe Sites of Colorado RFETS Special Nuclear Materials Program Building 371 P.O. Box 464 Golden, CO 80402-0464 Phone: 303/966-7336 Fax: 303/966-2241

Table A.24 (continued)

Site	Person(s) responsible for information/data	Person(s) responsible for material management
SRS	Thomas F. Severynse Westinghouse Savannah River Co. Nuclear Materials Stabilization Program Building 704-F Aiken, SC 29802 Phone: 803/952-4632 Fax: 803/952-4302	(Not applicable)
Y-12	Roger D. Keck Lockheed Martin Energy Systems, Inc. Nuclear Materials Management and Storage P.O. Box 2009 Building 9113 Mail Stop 8207 Oak Ridge, TN 37831-8207 Phone: 423/574-2525 Fax: 423/574-9811	Roger D. Keck
	NRC-licensed sites	
COMED3	Robert E. Berdelle Commonwealth Edison Company Comptroller Staff 36FN West P.O. Box 767 Chicago, IL 60690-0767 Phone: 312/394-2942 Fax: 312/394-2954	Michael J. Wallace Commonwealth Edison Company 37 FN West P.O. Box 767 Chicago, IL 60690-0767 Phone: 312/394-4200 Fax: 312/394-XXXX
FSVR	Les C. Hutchins Nuclear Decommissioning Public Service Company of Colorado Fort St. Vrain Reactor Site 16805 WCR 19 1/2 Platteville, CO 80651-9298 Phone: 303/620-1203 Fax: 303/620-1707	Ted Borst Nuclear Decommissioning Public Service Company of Colorado Fort St. Vrain Reactor Site 16805 WCR 19 1/2 Platteville, CO 80651-9298 Phone: 303/620-1000 Fax: 303/620-1707
GA	Chester L. Wisham General Atomics Nuclear Material Accountability Mail Stop 15-201 3550 General Atomics Court P.O. Box 85608 San Diego, CA 92121-1194 Phone: 619/455-4171 Fax: 619/455-2822	Chester L. Wisham

Appendix B. CHARACTERISTICS OF UNIRRADIATED ^{233}U REACTOR FUEL AT INEEL

B.1 INTRODUCTION

As indicated in Table A.2 of Appendix A, a major component of the domestic ^{233}U inventory is the unirradiated light-water breeder reactor (LWBR) fuel currently stored at the Idaho National Engineering and Environmental Laboratory (INEEL). As part of the U.S. Atomic Energy Commission—Energy Research and Development Administration (AEC—ERDA) LWBR Development Program, this fuel was originally fabricated for refueling the LWBR core of the government-owned reactor at the Shippingport Atomic Power Station. However, before the fuel could be used in the Shippingport LWBR core, the LWBR Development Program was terminated, and the Shippingport reactor was subsequently dismantled.

The unirradiated LWBR fuel from Shippingport was brought to INEEL originally for storage. There are no plans to use this fuel, and current plans are to continue storing and monitoring this material until it can be shipped to a spent nuclear fuel (SNF) or high-level waste (HLW) repository (DOE Idaho Operations Office July 19, 1995). This appendix describes the specific characteristics of this unirradiated fuel, and much of the information presented is based on Detrick (May 6, 1997 and Apr. 8, 1998) and the LWBR fuel *Final Safety Analysis Report* (FSAR) [Bolton, Christensen, and Hallinan (March 1989)].

B.2 LWBR HISTORY AND OPERATION

The LWBR core in the Shippingport reactor was a uniquely designed seed-blanket type, as shown in Fig. B.1 [Connors et al. (January 1979)]. This core operated in the Shippingport Power Station in Pennsylvania from 1977 to 1982, generating a gross electrical output of 2,128,943,470 kWh [Atherton (Oct. 1987)].

As indicated in Fig. B.2 [DiGuiseppe and Johnson (July 1982)], the LWBR core consisted of 12 "seed" fuel assemblies—hexagonal modules arranged in a symmetrical array, surrounded by a reflector-blanket region. Each module contained an axially movable "seed" region [which had a multiplication factor (k) greater than unity], and a stationary, annular hexagonal blanket (which had $k < 1$). Each of these regions, in turn, consisted of arrays of tightly packed, but not touching, fuel rods, which contained pellets of ThO_2 (thoria) and $^{233}UO_2$ (urania), the latter in varying amounts from 0 to 6 wt % in the seed and from 0 to 3 wt % in the blanket region [Lamarsh (1975)].

Figure B.3 [Bolton, Christensen, and Hallinan (March 1989)] gives a cutaway view of an LWBR seed module, and a similar view for an LWBR blanket module is provided in Fig. B.4. The seed-blanket module combination provided a unique binary (ThO_2 and UO_2) fuel control and distribution scheme, as shown in Figs. B.5 [Connors et al. (January 1979)] and B.6 [Heckler (June 1979)].

B.3 SHIPMENT HISTORY OF UNIRRADIATED FUEL

Shipments of unirradiated LWBR fuel to the ICPP for storage began in December 1984 and continued until December 1985. Forty unirradiated fuel canisters were shipped in 10 shipments (4 canisters per shipment). In June 1987, one unirradiated LWBR seed module was shipped from Bettis Atomic Power Laboratory (BAPL) to the ICPP along with a shipment of irradiated fuel.

The unirradiated fuel was shipped directly from BAPL to ICPP while the irradiated fuel from BAPL was shipped to the INEEL Naval Reactors Facility (NRF) for repackaging and eventual transfer to the ICPP Irradiated Fuel Storage Area. Layouts of the LWBR Fuel Storage Facility at the INEEL ICPP (for both irradiated and unirradiated fuels) are given in Figs. B.7 and B.8 [both from Bolton, Christensen, and Hallinan (March 1989)]. At the ICPP, contents of the unirradiated fuel canisters from BAPL were not reopened and inspected. The tamper-indicating devices on these canisters were left sealed. The sealed canisters were received at the Unirradiated LWBR storage area and placed directly in 20 of the 22 dry storage vaults of that facility. The unirradiated module, which came later, was placed inside a liner in a dry storage vault (labeled U-22) at the southern end of the unirradiated LWBR Fuel Storage Area.

B.4 MATERIAL CHARACTERISTICS

B.4.1 Storage Location and Facility

The unirradiated fuel is stored in underground, dry storage vaults which are located at the INEEL—ICPP Dry Vault Area (Facility 749). The Facility 749 area is an underground, dry storage facility for reactor fuel. Each unirradiated fuel storage dry vault is a shallow-lined hole sized to hold two fuel storage canisters end to end and is designed to isolate the fuel. A total of 22 dry storage vaults (labeled U-1 through U-22) were built to store forty 8-5/8-in.-diam unirradiated fuel storage canisters, to store one spare seed module, and to maintain one spare storage space.

B.4.2 Storage Containers

Each position in the Facility 749 area storing the unirradiated LWBR fuel contains unirradiated fuel rods, which are stored in outer stainless steel (SS) containers or canisters [see Fig. B.9, which is based on Bolton, Christensen, and Hallinan (March 1989)]. Each canister is constructed from 8-in.-schedule SS pipe and is called a fuel-handling unit (FHU). The unirradiated fuel rods are stored in a total of 40 canisters that fill 20 vault positions. In addition, one spare unirradiated seed module, contained in a liner, is stored in an additional vault (U-22) in the unirradiated LWBR fuel storage area.

B.4.3 Stored Material Characteristics

The unirradiated LWBR fuel inventory consists of the following: (1) rods not used in the fabrication of the Shippingport reactor core, (2) rods used for the breeder mock-up (BMU) tests, (3) rods used in preliminary hot-cell-criticality experiments that were part of the LWBR development program, and (4) loose pellets stored in SS tubes, which were placed in the unirradiated storage canisters. Table B.1 shows the total material inventory and describes the unirradiated rods shipped from BAPL to Idaho National Engineering Laboratory (now INEEL). The quantity of ^{233}U per (unirradiated) rod depends on (1) whether the rod type is a seed (S), standard-blanket (B), or power-flattening blanket (PFB) and (2) whether the rods were used in critical experiments or in the reactor core. Major characteristics of the storage containers are provided in Table B.2.

The LWBR fuel consists of ceramic pellets made either of ThO_2 (thoria) or of varying amounts of fissile uranium oxide (urania) (UO_2) mixed with ThO_2. Both ThO_2 and UO_2 melt above 5000°F, are insoluble in water, and have densities between 9 and 10 g/cm^3. The ceramic pellets are stacked in long tubes that are fabricated from Zircaloy-4.

A total of 14,468 rods of unirradiated fuel are stored in the 40 SS canisters described previously. This quantity is based on a BAPL review made of the backup information that was included along with the shipping receipts (DOE 741 forms) accompanying the unirradiated materials shipped. As indicated in Tables B.1 and B.2, the rods consist of a variety of types and include: aluminum spacer tubes, core retainer rods, mock-up blanket rods, mock-up blanket seed rods, PFB rods, retainer-blend rods (pellet tubes), scrap-blend rods (pellet tubes), seed rods, standard-blanket rods, and miscellaneous rods.

Table B.3 [based on Detrick (May 6, 1997, and Apr. 8, 1998) and Sadler (June 10, 1997)] gives the dry storage vault locations and summary ^{233}U contents of the BAPL container shipments of the unirradiated LWBR fuel to the INEEL—ICPP. For completeness, the major characteristics of the unirradiated seed module shipped with the irradiated fuel are given in Table B.4. The ^{232}U content associated with the ^{233}U inventory of the dry storage vaults is 5–10 ppm. The quantity of ^{233}U per fuel rod depends on the type of rod (seed, standard blanket, or PFB) and its intended use (reactor core or criticality experiments). Different rod types are found in each of the 40 SS canisters stored in Facility 749. Very little void space is found in any storage canister because after a storage canister was loaded with fuel rod inventory, most of the remaining void volume was filled with aluminum rods. Detailed ^{233}U characteristics of the unirradiated LWBR fuel stored in each of the INEEL—ICPP dry storage vaults (U-1 through U-22) are provided in Tables B.5.U1 through B.5.U22, respectively. (Note: There is no table for dry storage vault U-21, which is empty.) Each table lists the inventory characteristics of each shipping canister stored in a particular dry storage vault. These characteristics are based mainly on a detailed review by BAPL of the shipment records prepared for each canister of unirradiated fuel before it was shipped from BAPL to INEEL [Detrick (Apr. 8, 1998)]. The tables are arranged such that, in general, two LWBR canisters inside an ICPP dry storage vault are grouped together (i.e., stacked) under the vault number. The unirradiated LWBR fuel rods described in Tables B.5.U1 through B.5.U20 fall into four major groups: (1) production fuel; (2) detailed cell fuel; (3) breeder mockup fuel; and (4) other fuel, including proof of breeding (POB) fuel and miscellaneous.

The design of an unirradiated fuel storage canister is shown in the upper half of Fig. B.10 [Bolton, Christensen, and Hallinan (March 1989)]. It is constructed of 8-5/8-in.-OD × 7-5/8-in.-ID SS pipe. The interior length is 122.5 in., a length which allows for storage of the Shippingport LWBR fuel rods and those rods used in criticality experiments that were part of the LWBR Program. The bottom plate of the canister is constructed of 1/4-in.-thick SS plate and is welded to the canister wall. The top cover has a plug fitting into the canister, which is connected to a flange to seat it over the canister top. The plug section is made of a 7-5/8-in.-diam, 2-in.-thick SS plate welded to a 1/4-in.-thick SS flange with an 8-5/8-in.-diam. The 1/4-in. plate fits on the canister, whereas the 2-in.-thick section fits inside the canister. The cover and canister are connected by four 3/8-in., self-locking screws which screw in flush with the canister at right angles on the sides. The top center of the cover contains a threaded hole to allow handling of the canister using eyebolts or lifting rods. The canister lid fits tightly to the canister but does not provide an environmental seal.

Shorter BMU fuel rods are packaged into smaller inner canisters, which are loaded into the storage canister previously described. As shown in the lower half of Fig. B.10, each inner canister is constructed like the larger storage canister—except that 7-1/2-in.-OD by 7-in.-ID SS pipe is used, and the top cover is only 1-in. thick. The inner canisters have internal lengths of either 29-1/4 in. or 43-3/4 in. for BMU seed and blanket-rod storage, respectively. Two inner canisters containing 42-in.-long BMU blanket rods and one inner canister containing 24-in.-long BMU seed rods are placed in each storage canister. This placement was done to average the uranium loading per storage canister and to prevent rods from sliding back and forth during handling. As with the larger storage canisters, aluminum rods were used to "fill up" the inner canisters after the BMU rods were loaded.

B.4.4 Planned Disposition

This material was originally brought to INEEL for storage. Current plans call for continued storage and monitoring of the material until it can be shipped to a SNF or a HLW repository. Currently, there are no plans to use this unirradiated LWBR fuel [U.S. DOE Idaho Operations Office (July 19, 1995)].

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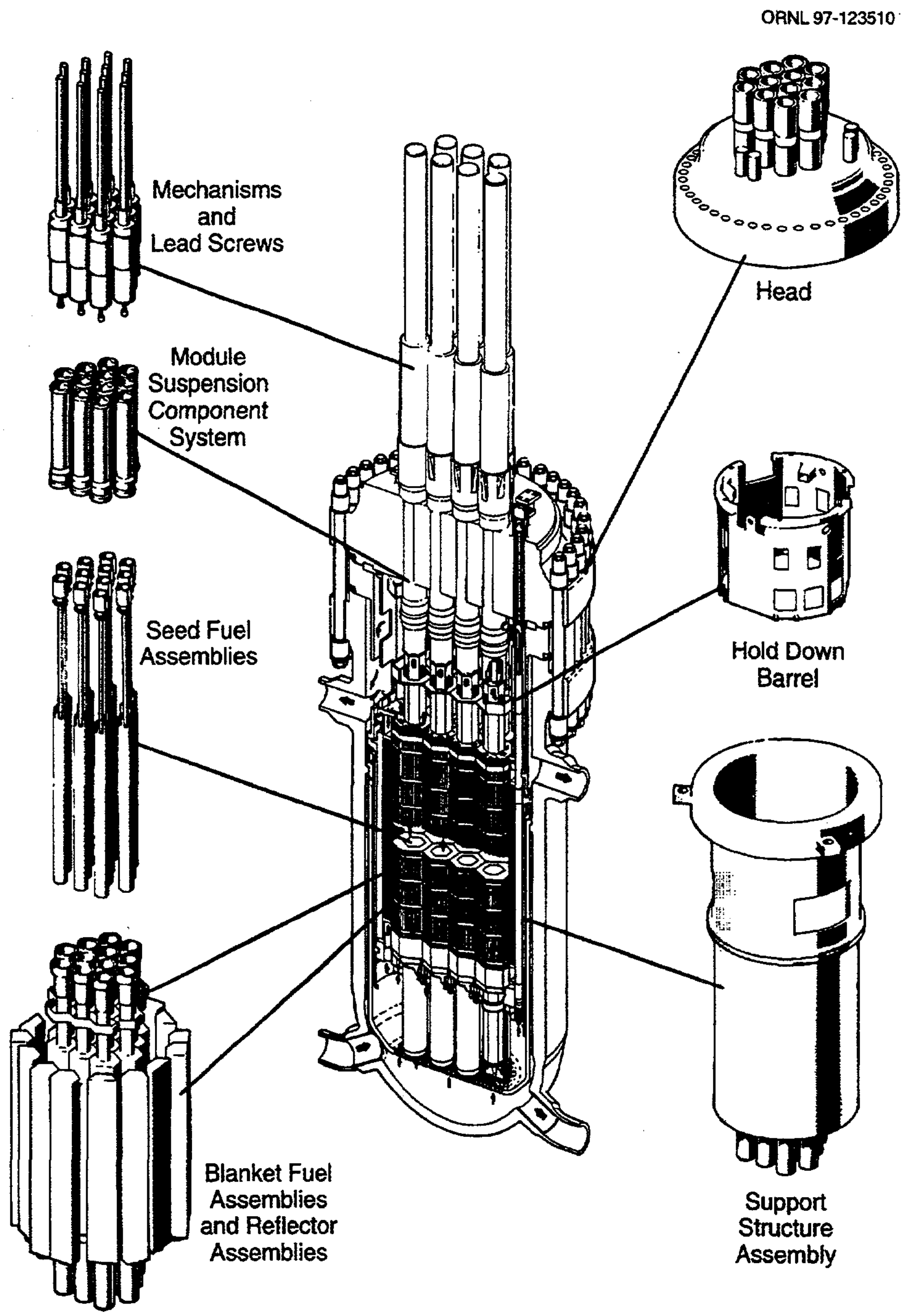


Fig. B.1. LWBR core in Shippingport reactor vessel. *Courtesy of Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.*

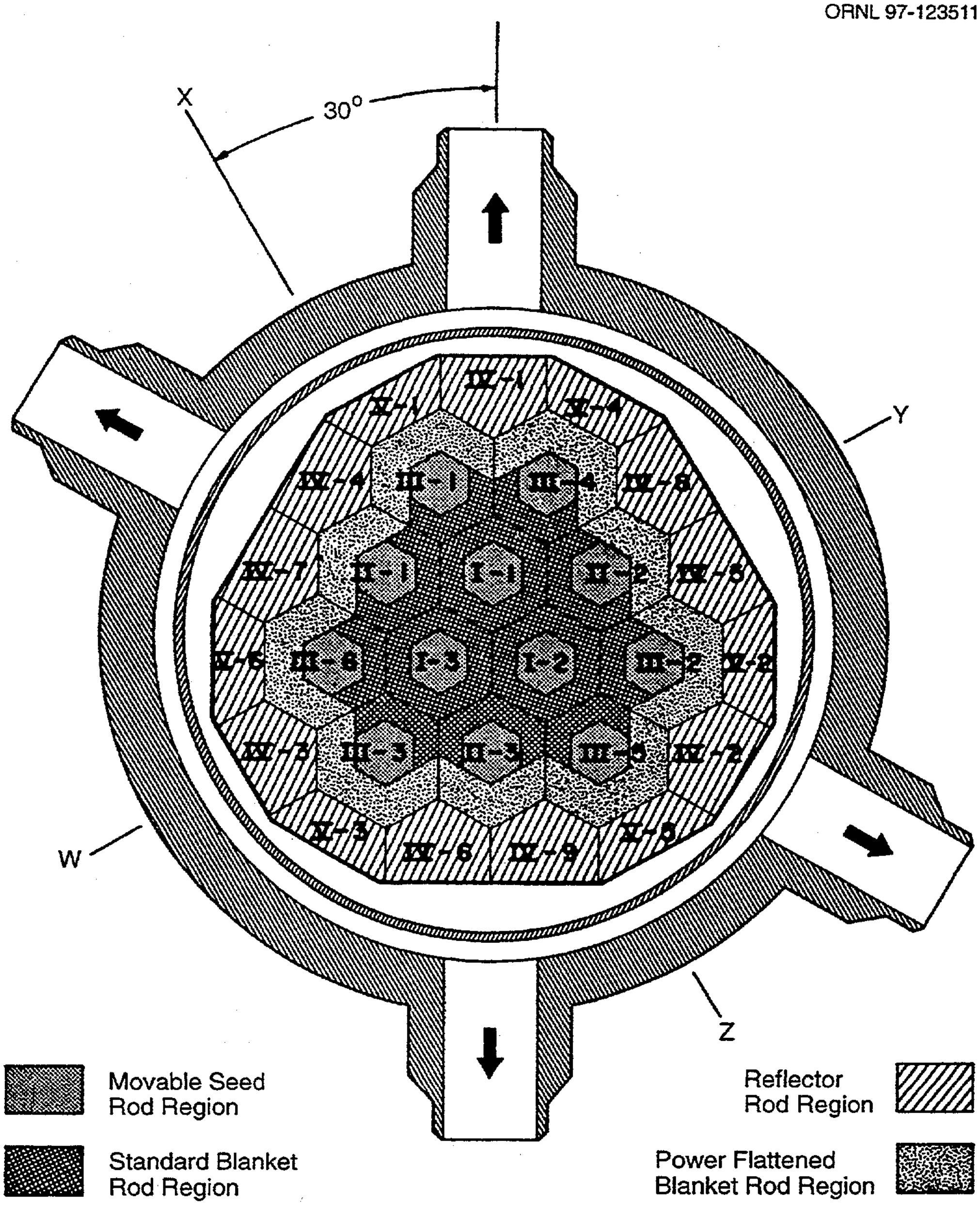


Fig. B.2. Cross-sectional diagram of LWBR core, showing module identification.
Courtesy of Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.

ORNL 97-123512

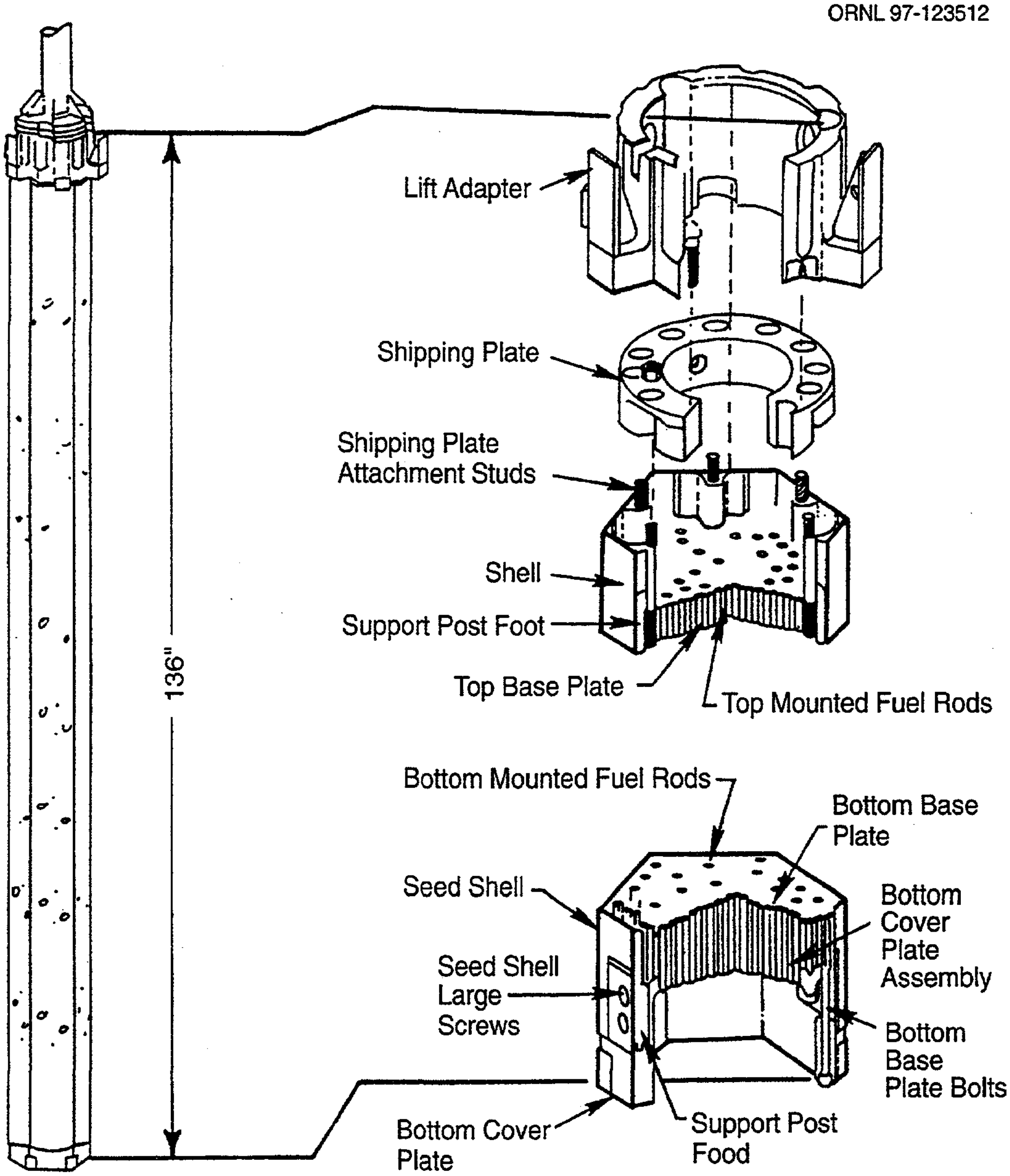


Fig. B.3. Cutaway view of an LWBR seed module. *Courtesy of Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.*

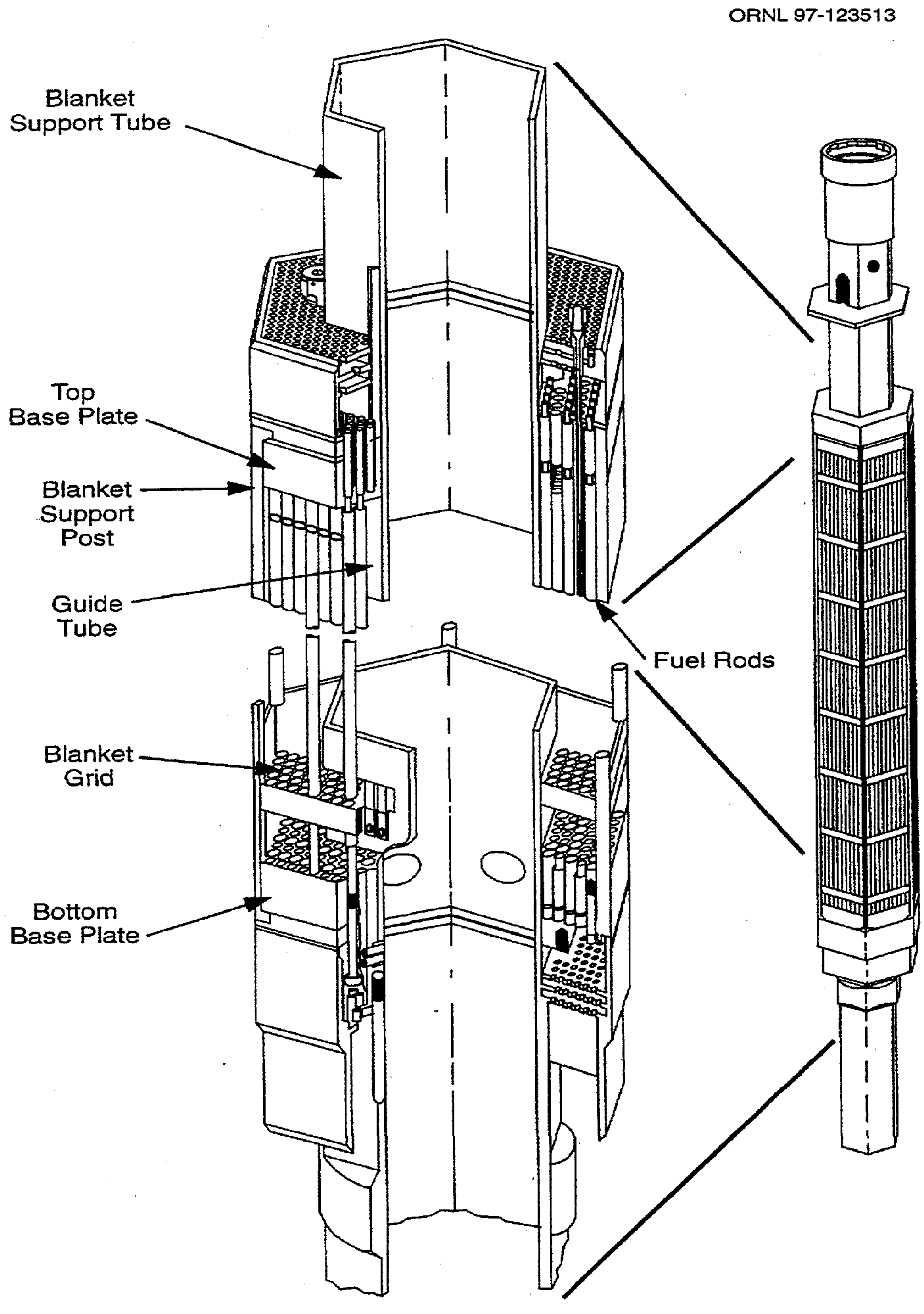


Fig. B.4. Shippingport LWBR blanket module.

ORNL 97-123514

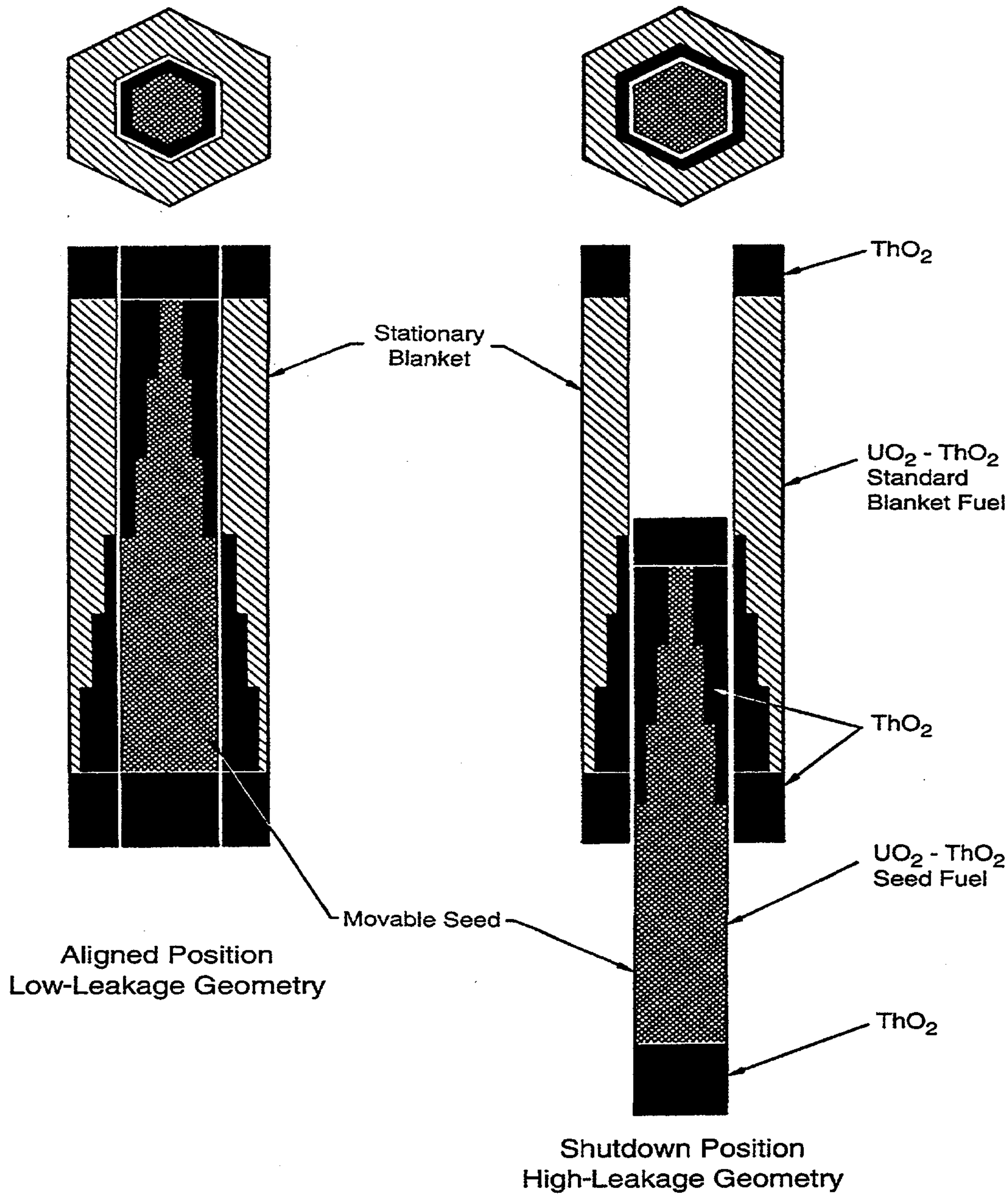


Fig. B.5. Vertical cross section of a typical LWBR seed-blanket-module combination, showing movable fuel control and "stepped" binary fuel distribution. *Courtesy of Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.*

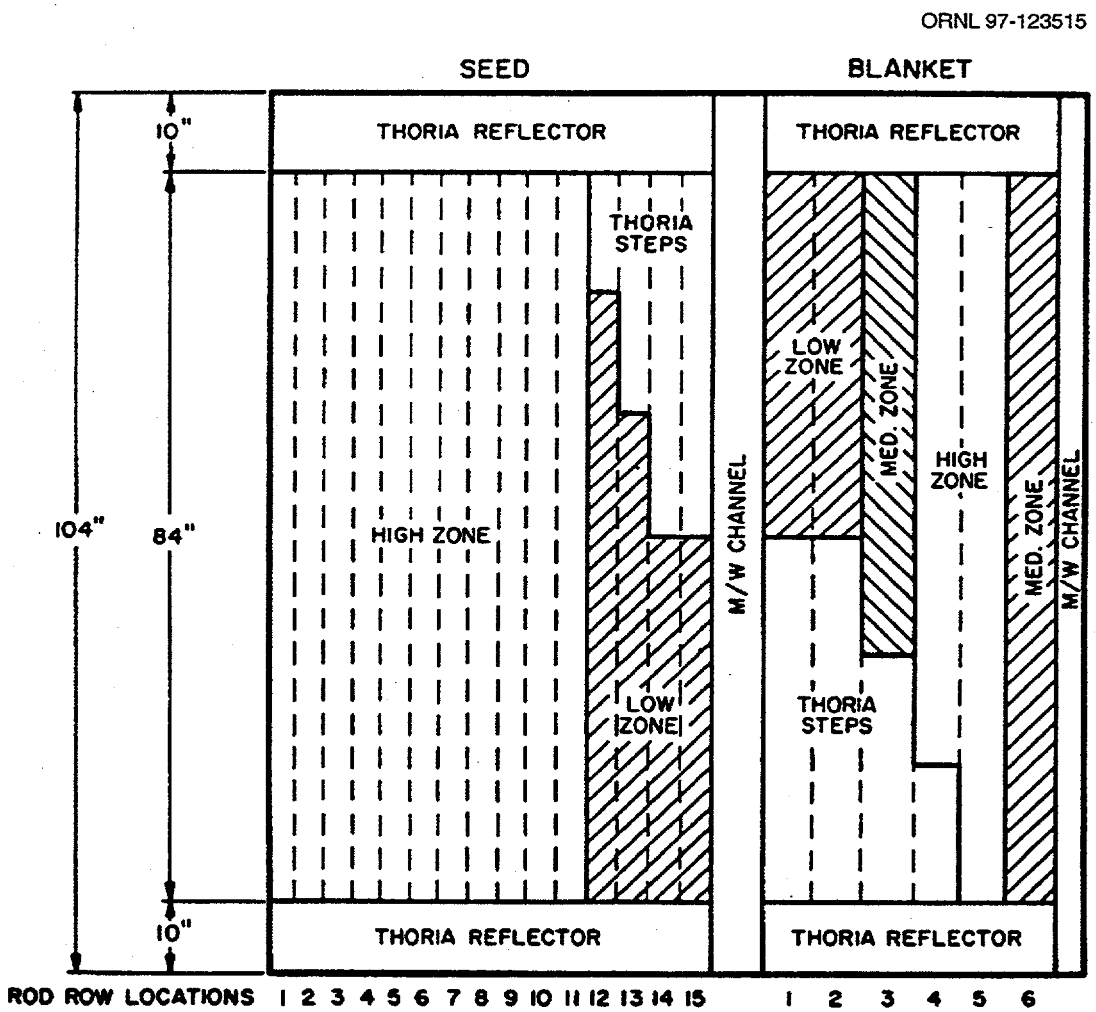


Fig. B.6. Fuel rod location schematic of a typical LWBR seed-blanket-module combination, showing radial and axial zoning of fuel. (The module centerline is at the left boundary.) *Courtesy of Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.*

ORNL 97-123516

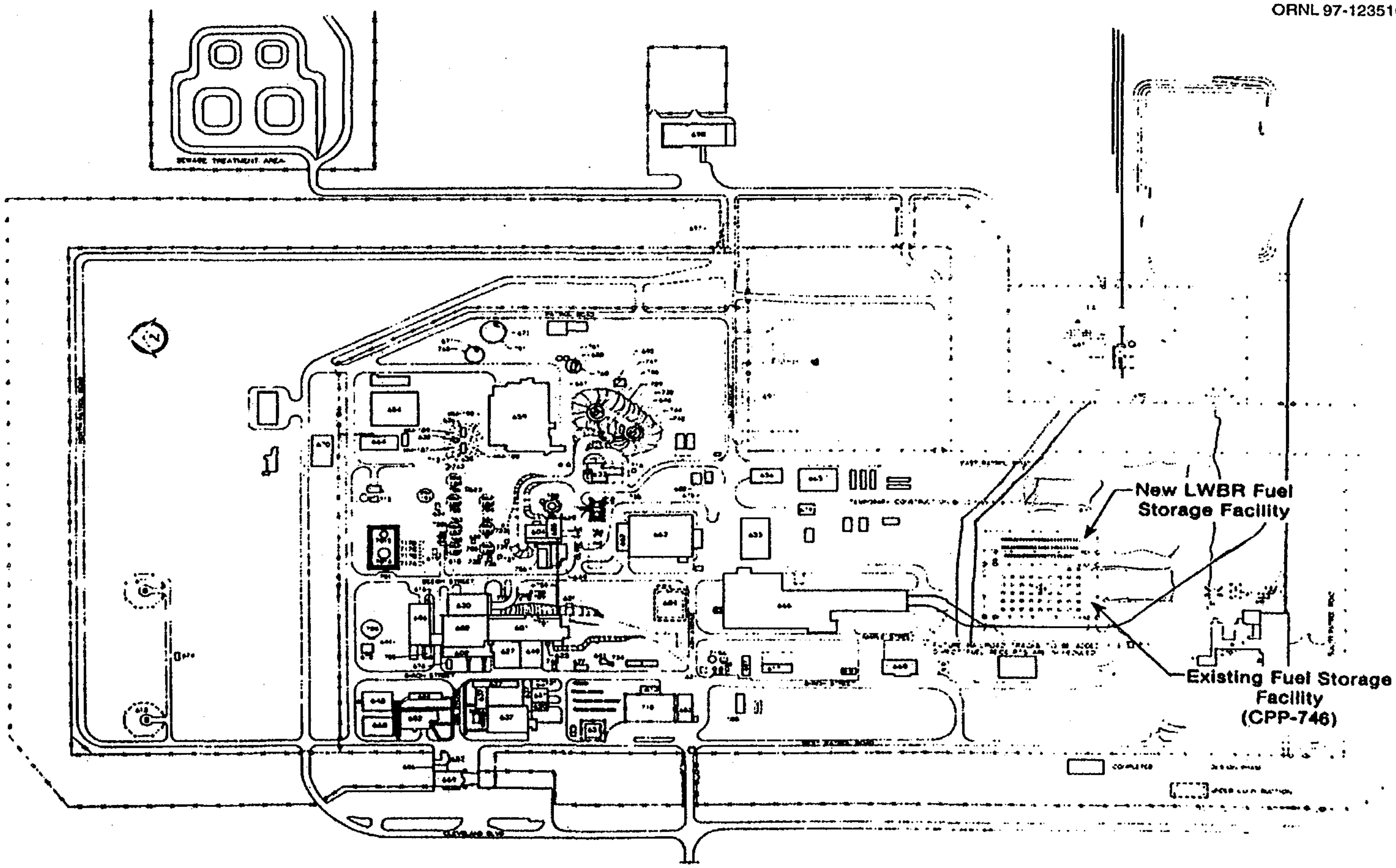


Fig. B.7. Layout of the LWBR fuel storage facility at the INEEL-ICPP. *Courtesy of Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.*

ORNL 97-123517

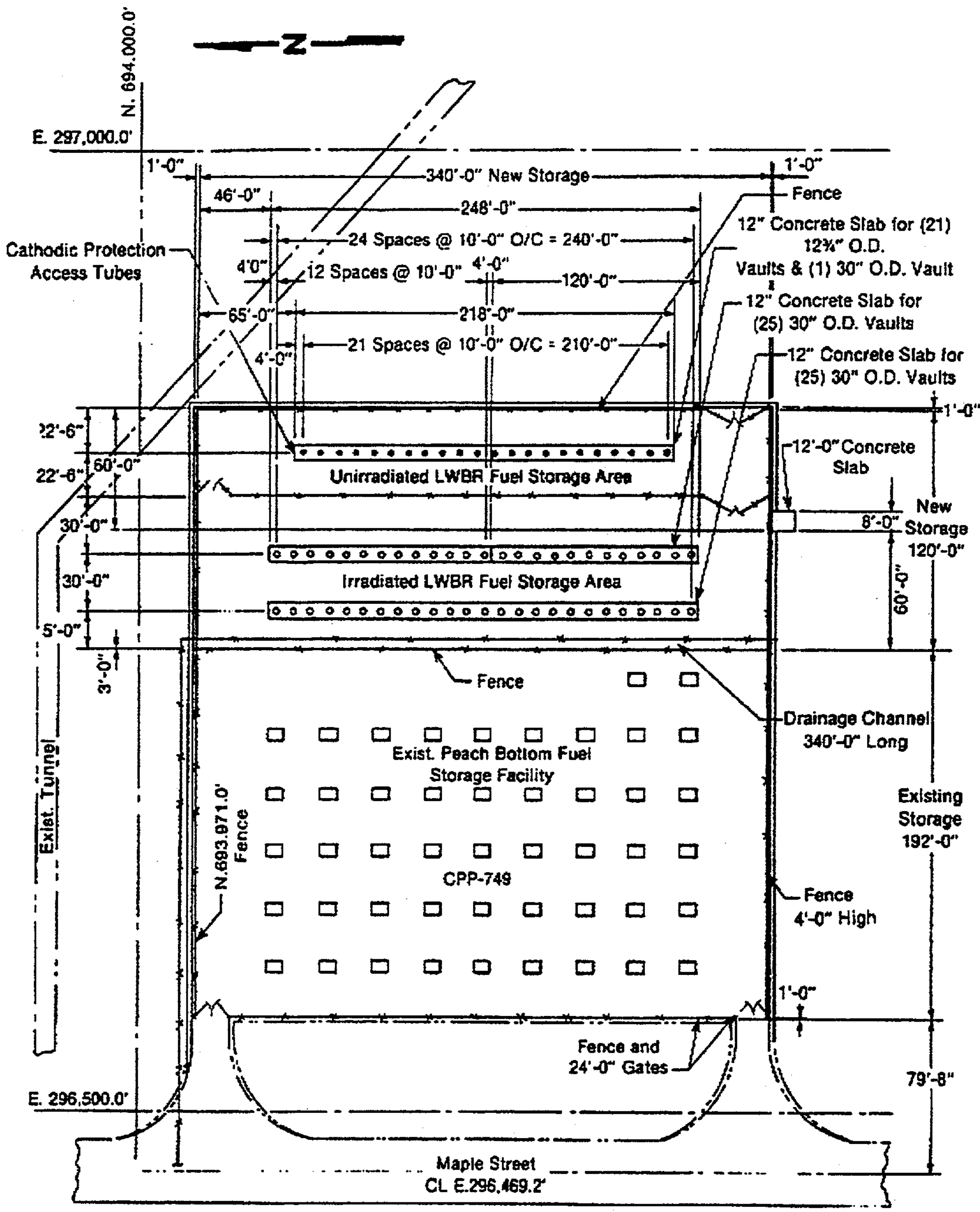
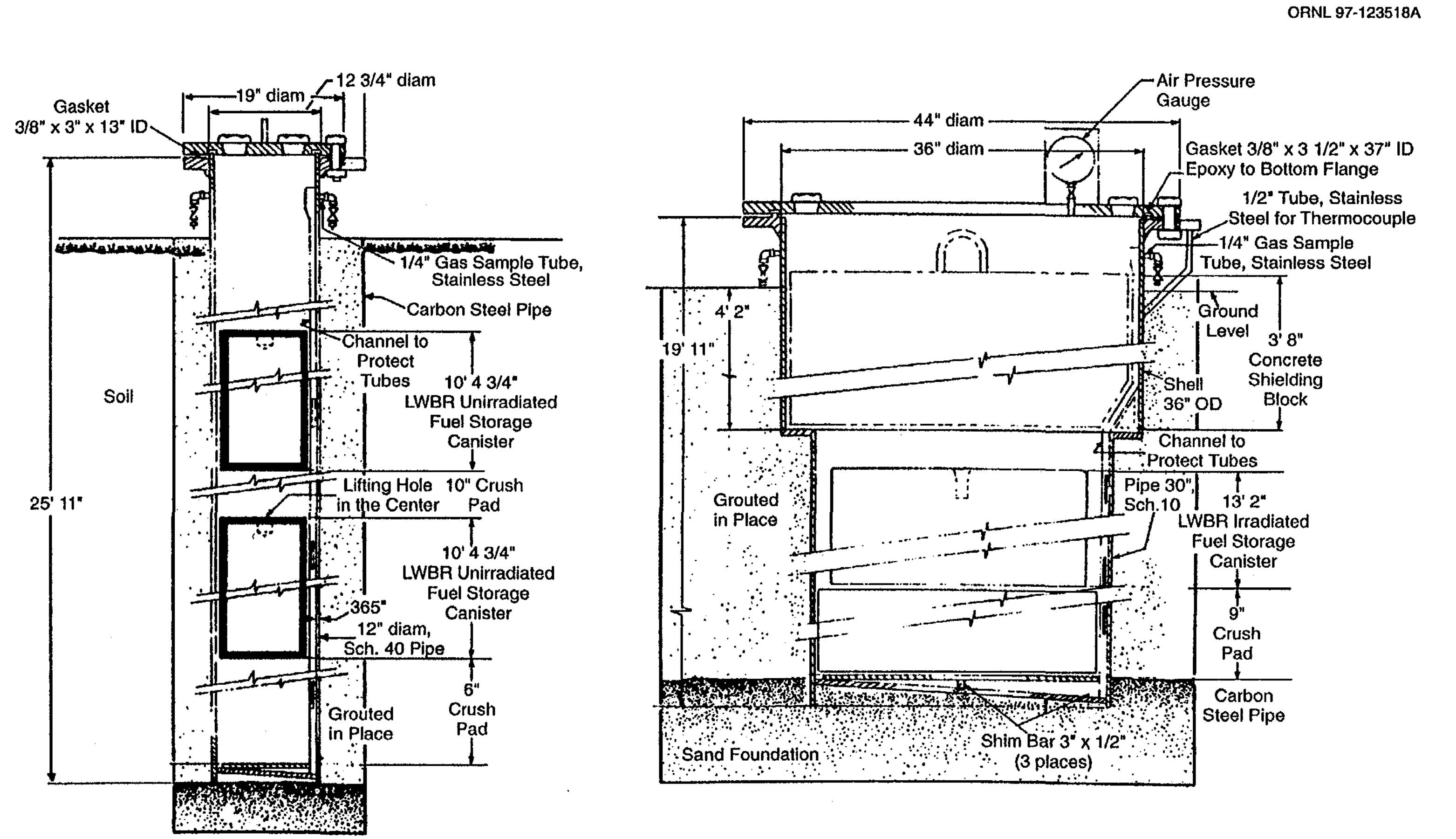


Fig. B.8. Plot plan of the LWBR fuel storage facility. *Courtesy of Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.*



a. Unirradiated LWBR Fuel Storage Dry Vault

b. Irradiated LWBR Fuel Storage Dry Vault

Fig. B.9. Section views of the dry-vault design for both unirradiated and irradiated LWBR fuel storage dry vaults at INEEL. The vault on the left stores the canisters of unirradiated LWBR fuel rods while the vault on the right stores both the irradiated LWBR fuel and the single unirradiated spare LWBR seed module. *Courtesy of Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.*

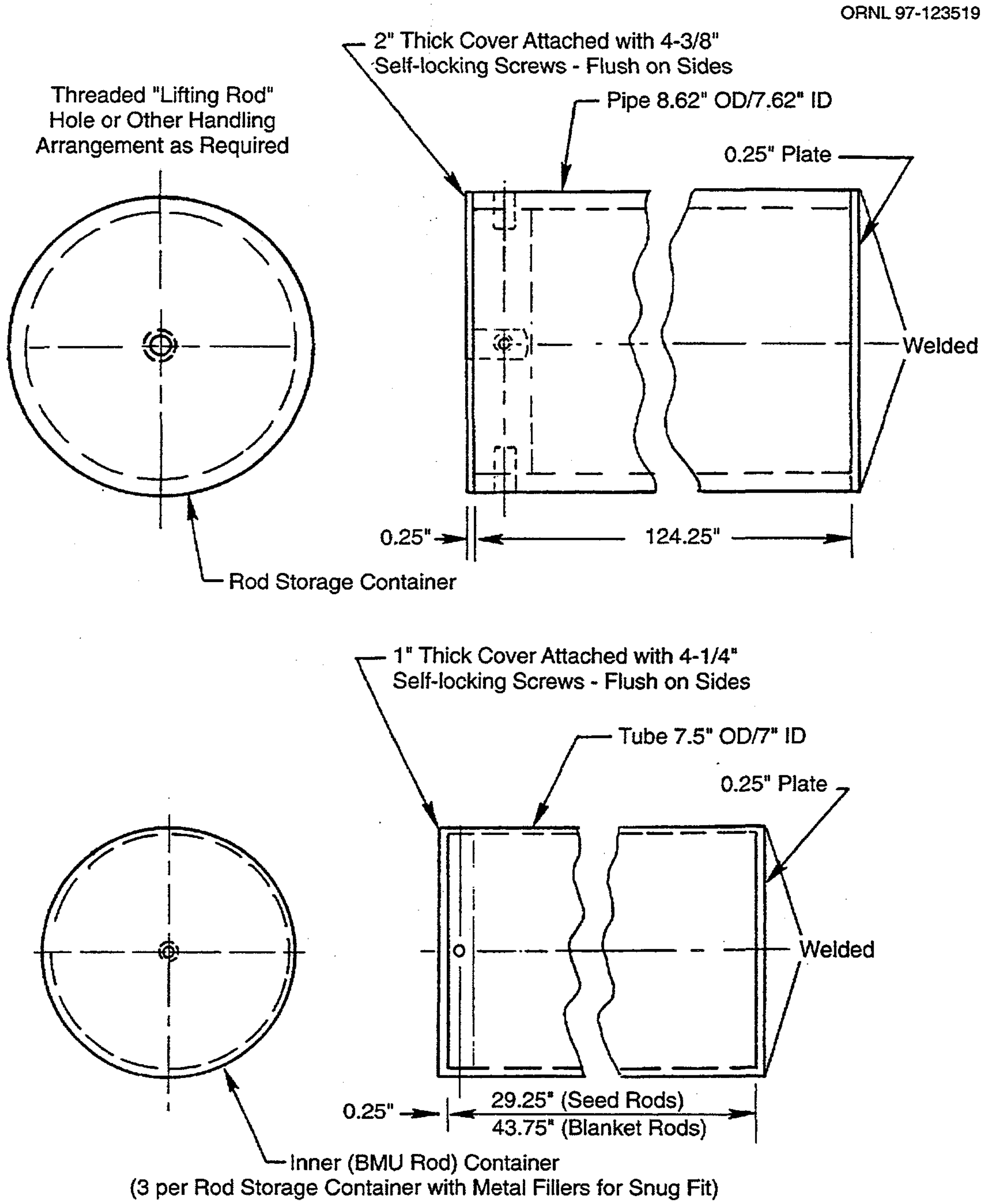


Fig. B.10. Axial and radial views of an unirradiated LWBR fuel storage canister.
Courtesy of Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.

Table B.1. Unirradiated LWBR fuel characteristics[a]

Rod type	Number of rods	^{233}U in fuel		Rod dimensions, in.			Rod masses, kg		
		wt %	kg	Outer diameter	Length	Clad thickness	Average cladding hardware	Fuel	Total
1. Core and criticality experiments									
A. Seed	2,290	4.3–5.2	55.70	0.306	101.1–119.1	0.027	0.29	0.84	1.14
B. Standard blanket	1,667	1.0–2.0	55.52	0.572	104.9–122.1	0.028–0.031	0.83	3.38	4.21
C. Power-flattening blanket	1,630	1.6–2.7	82.16	0.528	104.9–122.2	0.020–0.029	0.71	2.85	3.56
2. BMU seed	6,896	2.5–12.0	57.05	0.25	28.2	0.019	0.041	0.335	0.376
3. BMU blanket	1,759	2.0	40.97	0.624	42.5	0.0305	0.307	1.911–2.824	2.218–3.131
4. Scrap pellets	NA[b]	2.0	5.9						
5. Retainer pellets and rods	*c*	1.3	3.5						
Total	14,468[d]		300.8						

[a]Adapted from Bolton, Christensen, and Hallinan (March 1989).
[b]These pellets are stored in stainless steel (SS) tubes inside a canister.
[c]Uncertain.
[d]Includes 226 scrap and retainer rods.

Table B.2. Type, number features, and total number of canisters required for unirradiated LWBR storage containers[a]

Rod type	Total number of rods	Features: Number of 1/4-in. OD aluminum rods to fill	Features: Number of rods	Features: ^{233}U content (kg)	Features: Total mass (kg)	Total number of canisters required
1. Core and criticality experiments						
A. Seed	2,290	56	458	11.14	781	5
B. Standard blanket	1,667	18	140	4.63	827	12
C. Power-flattening blanket	1,630	19	163	8.22	824[b]	10
2. Breeder mock-up (BMU) seed	6,896	51	575	9.20[c]	974–1,108[d]	12
3. BMU blanket	1,759	328	147			
4. Scrap pellets	NA[e]	NA		7.5	604	1[f]
5. Retainer pellets and rods	g	NA		7.5	671	1[f]
Total	14,468[h]					40

[a]Adapted from Bolton, Christensen, and Hallinan (March 1989).
[b]Includes 223 kg for the storage canister itself.
[c]Assumes two BMU blanket canisters and one BMU seed canister containing 12 wt % ^{233}U per unirradiated fuel storage canister, which produces the highest uranium content for the canisters containing the BMU rods, although other storage canisters containing one BMU blanket canister and two BMU seed canisters with 2 to 5 wt % ^{233}U were also shipped.
[d]Includes 223 kg, 29 kg, and 79 kg for storage canister, one BMU seed canister, and two BMU blanket canisters, respectively.
[e]These pellets are stored in stainless steel tubes inside the canister.
[f]These are the same canister.
[g]Uncertain.
[h]Includes 226 scrap and retainer rods.

Table B.3. Storage locations and contents of BAPL container shipments of unirradiated LWBR fuel to the INEEL—ICPP[a]

ID of dry storage vault[b]	Position[c]	ID of stored shipping container	Rods Type(s)[d]	Rods Number[e]	^{233}U content (grams)	ID of dry storage vault[b]	Position[c]	ID of stored shipping container	Rods Type(s)[d]	Rods Number[e]	^{233}U content (grams)
U-1	Top	LWB-04	S, B	744	9,032	U-11	Top	LWB-25	S	476	11,934
	Bottom	LWB-11	S, B	749	11,832		Bottom	LWB-38	B, PFB	156	9,324
U-2	Top	LWB-05	S, B	746	9,066	U-12	Top	LWB-24	B, PFB	148	7,061
	Bottom	LWB-13	S, B	738	11,927		Bottom	LWB-17	B, PFB	143	6,056
U-3	Top	LWB-02	S, B	762	8,132	U-13	Top	LWB-31	B, PFB	149	6,709
	Bottom	LWB-14	S, B	742	11,993		Bottom	LWB-28	S, B, PFB	250	7,760
U-4	Top	LWB-01	S, B	738	9,003	U-14	Top	LWB-37	B, PFB	147	5,198
	Bottom	LWB-15	S, B	707	11,520		Bottom	LWB-32	B, PFB	151	4,299
U-5	Top	LWB-08	S, B	1,232	6,015	U-15	Top	LWB-18	S, B, PFB	243	7,562
	Bottom	LWB-03	S, B	747	5,349		Bottom	LWB-19	S, B	177	5,747
U-6	Top	LWB-16	B	136	4,371	U-16	Top	LWB-30	S, B, PFB	183	5,270
	Bottom	LWB-09	S, B	759	4,086		Bottom	LWB-21	S, B, PFB	271	8,459
U-7	Top	LWB-39	B, PFB	139	3,562	U-17	Top	LWB-26	S, B, PFB	180	6,097
	Bottom	LWB-36	PFB	159	8,768		Bottom	LWB-23	S, B, PFB	263	10,114
U-8	Top	LWB-35	S, PFB	261	8,572	U-18	Top	LWB-33	S, B, PFB	178	6,464
	Bottom	LWB-34	S	462	10,493		Bottom	LWB-29	S, B, PFB	185	5,360
U-9	Top	LWB-07	B, PFB	149	7,438	U-19	Top	LWB-22	S, B, PFB	346	8,293
	Bottom	LWB-06	B, PFB	155	8,074		Bottom	LWB-20	B, PFB	150	7,401
U-10	Top	LWB-40	B	134	5,500	U-20	Top	LWB-42	S, B, PFB, R, M	122	3,339
	Bottom	LWB-27	B, PFB	153	6,712		Bottom	LWB-41	M	138	6,909
									Total	14,468	300,801

[a]Based on Detrick (May 6, 1997, and Apr. 8, 1998) and Sadler (June 10, 1997).
[b]Located at the INEEL—ICPP Unirradiated LWBR Fuel Storage Area (Facility 749).
[c]Each dry storage vault has space for two storage containers, one stacked on top of the other.
[d]Types of rods: seed (S), standard blanket (B), power-flattening blanket (PFB), reflector (R), and miscellaneous (M).
[e]Total number of rods (all types) in container (canister).

Table B.4. Characteristics of unirradiated spare LWBR seed module[a,b]

Category	Mass (kg)
Fuel module[c]	
Total U	16.84
^{233}U	16.56[d]
^{235}U	0
Thorium	434
Zircaloy-4	219
Ni-Cr-Fe alloy	42
Stainless steel	13
Fixtures	37
Fuel module total	762
Container	2,357
Storage total	3,119

[a]Based on Bolton, Christensen, and Hallinan (March 1989); Detrick (May 6, 1997); and Liable (Apr. 15, 1997).
[b]Stored in a single Type D canister.
[c]Contains 619 fuel rods.
[d]Associated ^{232}U content is 8 ppm (of total uranium).

Table B.5.U1. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-1[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Breeder mockup fuel				
LWB–04	Seed	Type P/O	5.0/9.0	567	4,529	4,436	60,300
	Standard blanket	Type E	2.0	177	4,693	4,596	230,500
	LWB–04 total			**744**	**9,222**	**9,032**	**290,800**
LWB–11	Seed	Type P/O	5.0/9.0	50	399	391	5,300
		Type S	12.0	523	7,015	6,871	51,900
	Subtotal			573	7,414	7,262	57,200
	Standard blanket	Type E	2.0	176	4,666	4,570	229,200
	LWB–11 total			**749**	**12,080**	**11,832**	**286,400**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for breeder mockup fuel rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

Table B.5.U2. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-2[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Breeder mockup fuel				
LWB–05	Seed	Type P/O	5.0/9.0	568	4,537	4,444	60,400
	Standard blanket	Type E	2.0	178	4,719	4,622	231,800
	LWB–05 total			**746**	**9,256**	**9,066**	**292,200**
LWB–13	Seed	Type S	12.0	564	7,565	7,409	56,000
	Standard blanket	Type E	2.0	174	4,613	4,518	226,600
	LWB–13 total			**738**	**12,178**	**11,927**	**282,600**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for breeder mockup fuel rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

Table B.5.U3. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-3[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Breeder mockup fuel				
LWB–02	Seed	Type P/O	5.0/9.0	584	4,665	4,569	62,100
	Standard blanket	Type E-1/E-2	2.0	61	536	525	80,500
		Type E	2.0	117	3,102	3,038	152,400
	Subtotal			178	3,638	3,563	232,900
	LWB–02 total			**762**	**8,303**	**8,132**	**295,000**
LWB–14	Seed	Type S	12.0	567	7,605	7,449	56,300
	Standard blanket	Type E	2.0	175	4,640	4,544	227,900
	LWB–14 total			**742**	**12,245**	**11,993**	**284,200**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for breeder mockup fuel rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

Table B.5.U4. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-4[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Breeder mockup fuel				
LWB–01	Seed	Type P/O	5.0/9.0	560	4,473	4,381	59,600
	Standard blanket	Type E	2.0	178	4,719	4,622	231,800
	LWB–01 total			**738**	**9,192**	**9,003**	**291,400**
LWB–15	Seed	Type S	12.0	533	7,149	7,002	52,900
	Standard blanket	Type E	2.0	174	4,613	4,518	226,600
	LWB–15 total			**707**	**11,762**	**11,520**	**279,500**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for breeder mockup fuel rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

Table B.5.U5. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-5[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Breeder mockup fuel				
LWB–08	Seed	Type M/N	2.5	1,143	4,558	4,465	126,000
	Standard blanket	Type E-3/E-4	2.0	89	1,583	1,550	116,900
	LWB–08 total			**1,232**	**6,141**	**6,015**	**242,900**
LWB–03	Seed	Type M/N	2.5	371	1,480	1,449	40,900
		Type P/O	5.0/9.0	199	1,589	1,557	21,200
	Subtotal			570	3,069	3,006	62,100
	Standard blanket	Type E-1/E-2	2.0	84	738	723	110,900
		Type E-3/E-4	2.0	93	1,654	1,620	122,200
	Subtotal			177	2,392	2,343	233,100
	LWB–03 total			**747**	**5,461**	**5,349**	**295,200**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for breeder mockup fuel rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

Table B.5.U6. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-6[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–16	Standard blanket	H-84	2.0	24	1,335	1,311	70,000
		H-70	2.0	25	1,156	1,134	73,000
		M-84	1.7	15	697	684	44,000
		M-56	1.7	4	123	121	12,000
		L-42	1.2	68	1,139	1,121	200,000
	LWB–16 total			**136**	**4,450**	**4,371**	**399,000**
			Breeder mockup fuel				
LWB–09	Seed	Type M/N	2.5	676	2,696	2,641	74,600
	Standard blanket	Type E-3/E-4	2.0	83	1,476	1,445	109,000
	LWB–09 total			**759**	**4,172**	**4,086**	**183,600**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches. Rod stratum not available for breeder mockup fuel rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by $[^{233}U\ (g) / (UO_2 + ThO_2)\ (g)] \times 100\%$. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

Table B.5.U7. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-7[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g Total U	Mass, g ^{233}U	Mass, g Th
			Detailed cell fuel				
LWB–39	Standard blanket	Type 1	0.963	80	1,011	977	222,700
		Type 2	1.532	24	645	632	66,200
	Subtotal			104	1,656	1,609	288,900
	PFB[e]	Type D	2.564	35	1,988	1,953	80,800
	LWB–39 total			**139**	**3,644**	**3,562**	**369,700**
LWB–36	PFB	Type C	2.564	1	48	47	2,300
		Type D	2.564	151	8,579	8,424	348,600
		Type E	1.928	7	303	297	16,400
	LWB–36 total			**159**	**8,930**	**8,768**	**367,300**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for detailed cell rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U8. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-8[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Detailed cell fuel				
LWB–35	Seed	Type 1	5.6	157	5,657	5,550	102,500
	PFB[e]	Type A	1.542	45	779	764	104,900
		Type B	1.928	28	812	798	65,800
		Type C	2.564	31	1,486	1,460	71,600
	Subtotal			104	3,077	3,022	242,300
	LWB–35 total			**261**	**8,734**	**8,572**	**344,800**
LWB–34	Seed	Type 1	5.6	174	6,270	6,151	113,600
		Type 2	3.8	66	1,359	1,335	44,500
		Type 3	3.8	72	1,190	1,170	49,100
		Type 4	3.8	150	1,872	1,837	102,900
	LWB–34 total			**462**	**10,691**	**10,493**	**310,100**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for detailed cell rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U9. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-9[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g Total U	Mass, g ^{233}U	Mass, g Th
			Production fuel				
LWB–07	Standard blanket	H-84	2.0	8	444	436	23,400
		H-70	2.0	26	1,202	1,182	75,600
		M-84	1.7	24	1,113	1,095	69,900
		M-56	1.7	8	246	242	23,600
		L-42	1.2	16	268	263	47,500
	Subtotal			82	3,273	3,218	240,000
	PFB[e]	H-84	2.6	67	4,303	4,220	163,700
	LWB–07 total			**149**	**7,576**	**7,438**	**403,700**
LWB–06	Standard blanket	H-84	2.0	8	444	436	23,400
		H-70	2.0	6	278	273	17,400
		M-84	1.7	8	371	365	23,300
		M-56	1.7	8	246	242	23,600
		L-42	1.2	8	134	131	23,700
	Subtotal			38	1,473	1,447	111,400
	PFB	H-84	2.6	93	5,972	5,859	227,600
		M-84	1.9	8	378	371	19,600
		M-56	1.9	8	250	246	19,900
		L-42	1.5	8	154	151	19,800
	Subtotal			117	6,754	6,627	286,900
	LWB–06 total			**155**	**8,227**	**8,074**	**398,300**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U10. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-10[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Detailed cell fuel				
LWB–40	Standard blanket	Type 2	1.532	19	510	501	52,400
		Type 3	1.901	45	1,870	1,814	123,800
		Type 4	1.901	47	2,352	2,281	129,300
		Type 5	1.532	23	922	904	63,300
	LWB–40 total			**134**	**5,654**	**5,500**	**368,800**
			Production fuel				
LWB–27	Standard blanket	H-84	2.0	8	445	437	23,400
		H-70	2.0	6	277	273	17,400
		M-84	1.7	8	371	365	23,300
		M-56	1.7	8	246	242	23,600
		L-42	1.2	27	452	444	79,600
	Subtotal			57	1,791	1,761	167,300
	PFB[e]	H-84	2.6	48	3,080	3,022	117,700
		H-70	2.6	8	428	419	19,500
		M-84	1.9	24	1,134	1,113	58,800
		M-56	1.9	8	250	245	19,900
		L-42	1.5	8	154	152	19,800
	Subtotal			96	5,046	4,951	235,700
	LWB–27 total			**153**	**6,837**	**6,712**	**403,000**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches. Rod stratum not available for detailed cell rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U11. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-11[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–25	Seed	H-84	5.2	222	7,644	7,521	153,000
		L-70	4.2	32	779	766	22,500
		L-56	4.2	96	1,868	1,837	68,300
		L-42	4.2	126	1,840	1,810	90,600
	LWB–25 total			**476**	**12,131**	**11,934**	**334,400**
LWB–38	Standard blanket	H-84	2.0	16	890	874	46,800
		H-70	2.0	13	598	588	37,900
		M-84	1.7	8	371	365	23,300
	Subtotal			37	1,859	1,827	108,000
	PFB[e]	H-84	2.6	119	7,649	7,497	290,000
	LWB–38 total			**156**	**9,508**	**9,324**	**398,000**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U12. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-12[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–24	Standard blanket	H-84	2.0	33	1,796	1,764	96,600
		H-70	2.0	3	138	135	8,700
		M-84	1.7	8	372	365	23,300
		M-56	1.7	8	246	242	23,600
		L-42	1.2	32	536	527	94,500
	Subtotal			84	3,088	3,033	246,700
	PFB[e]	H-84	2.6	64	4,113	4,028	156,000
	LWB–24 total			**148**	**7,201**	**7,061**	**402,700**
LWB–17	Standard blanket	H-84	2.0	15	561	552	44,100
		H-70	2.0	8	370	364	23,300
		M-84	1.7	28	1,298	1,278	81,500
		M-56	1.7	20	616	605	59,100
		L-42	1.2	16	268	262	47,200
	Subtotal			87	3,113	3,061	255,200
	PFB	H-84	2.6	33	2,124	2,079	79,900
		H-70	2.6	7	375	368	17,100
		M-84	1.9	6	279	277	15,300
		M-56	1.9	7	219	215	17,400
		L-42	1.5	3	58	56	7,200
	Subtotal			56	3,055	2,995	136,900
	LWB–17 total			**143**	**6,168**	**6,056**	**392,100**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U13. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-13[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–31	Standard blanket	H-84	2.0	25	1,390	1,366	73,100
		H-70	2.0	10	463	454	29,100
		M-84	1.7	6	278	274	17,500
		M-56	1.7	16	494	486	47,300
		L-42	1.2	24	402	395	70,800
	Subtotal			81	3,027	2,975	237,800
	PFB[e]	H-84	2.6	54	3,466	3,399	131,300
		M-56	1.9	6	187	184	14,900
		L-42	1.5	8	154	151	20,000
	Subtotal			68	3,807	3,734	166,200
	LWB–31 total			**149**	**6,834**	**6,709**	**404,000**
LWB–28	Seed	H-84	5.2	45	1,573	1,547	31,100
		L-70	4.2	38	923	909	26,500
		L-56	4.2	25	487	478	17,800
		L-42	4.2	47	688	676	33,800
	Subtotal			155	3,671	3,610	109,200
	Standard blanket	H-84	2.0	8	444	437	23,500
		H-70	2.0	3	138	136	8,800
		M-84	1.7	8	371	364	23,300
		M-56	1.7	31	950	935	91,000
		L-42	1.2	12	202	197	35,400
	Subtotal			62	2,105	2,069	182,000
	PFB	H-84	2.6	33	2,121	2,081	81,300
	LWB–28 total			**250**	**7,897**	**7,760**	**372,500**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U14. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-14[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–37	Standard blanket	H-84	2.0	9	495	485	26,200
		H-70	2.0	26	1,045	1,024	71,700
		M-56	1.7	25	768	755	73,900
		L-42	1.2	17	281	276	50,000
	Subtotal			77	2,588	2,541	221,800
	PFB[e]	H-84	2.6	19	1,198	1,175	46,300
		H-70	2.6	10	525	514	24,100
		M-84	1.9	2	86	85	4,600
		M-56	1.9	13	402	394	32,100
		L-42	1.5	26	497	488	64,600
	Subtotal			70	2,708	2,656	171,700
	LWB–37 total			**147**	**5,297**	**5,198**	**393,500**
LWB–32	Standard blanket	H-84	2.0	1	56	55	2,900
		H-70	2.0	6	277	272	17,400
		M-84	1.7	6	278	273	17,500
		M-56	1.7	17	524	516	50,400
		L-42	1.2	15	251	248	44,400
	Subtotal			45	1,386	1,364	132,600
	PFB	H-84	2.6	2	128	126	4,800
		H-70	2.6	6	321	315	14,600
		M-84	1.9	12	568	558	29,600
		M-56	1.9	26	812	798	64,800
		L-42	1.5	60	1,156	1,138	149,000
	Subtotal			106	2,985	2,935	262,800
	LWB–32 total			**151**	**4,371**	**4,299**	**395,400**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / ($UO_2 + ThO_2$) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U15. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-15[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–18	Seed	H-84	5.2	60	2,108	2,073	41,500
		L-56	4.2	16	311	306	11,400
		L-42	4.2	64	932	917	46,000
	Subtotal			140	3,351	3,296	98,900
	Standard blanket	H-84	2.0	16	890	874	46,800
		M-84	1.7	10	465	456	29,100
		L-42	1.2	32	535	526	94,000
	Subtotal			58	1,890	1,856	169,900
	PFB[e]	H-84	2.6	24	1,544	1,514	58,700
		M-84	1.9	16	755	742	39,200
		M-56	1.9	5	156	154	12,500
	Subtotal			45	2,455	2,410	110,400
	LWB–18 total			**243**	**7,696**	**7,562**	**379,200**
LWB–19	Seed[f]	H-84	5.2	39	1,372	1,350	26,900
		L-70	4.2	7	167	164	4,900
		L-56	4.2	1	17	16	700
		L-42	4.2	13	183	181	9,300
	Subtotal			60	1,739	1,711	41,800
	Standard blanket	H-70	2.0	54	2,497	2,451	157,000
		M-84	1.7	15	696	684	43,700
		M-56	1.7	8	246	242	23,700
		L-42	1.2	40	670	659	118,100
	Subtotal			117	4,109	4,036	342,500
	LWB–19 total			**177**	**5,848**	**5,747**	**384,300**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

[f]Contains detailed seed rods.

Table B.5.U16. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-16[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–30	Seed	H-84	5.2	16	563	553	11,100
		L-56	4.2	16	311	306	11,400
		L-42	4.2	16	234	230	11,500
	Subtotal			48	1,108	1,089	34,000
	Standard blanket	H-84	2.0	16	888	873	46,800
		H-70	2.0	8	369	363	23,300
		M-84	1.7	8	372	366	23,300
		L-42	1.2	39	653	642	115,300
	Subtotal			71	2,282	2,244	208,700
	PFB[e]	H-70	2.6	16	858	840	39,000
		M-56	1.9	16	500	491	39,800
		L-42	1.5	32	616	606	79,500
	Subtotal			64	1,974	1,937	158,300
	LWB–30 total			**183**	**5,364**	**5,270**	**401,000**
LWB–21	Seed	H-84	5.2	85	2,982	2,934	58,500
		L-70	4.2	16	388	381	11,200
		L-56	4.2	19	369	363	13,500
		L-42	4.2	51	744	732	36,700
	Subtotal			171	4,483	4,410	119,900
	Standard blanket	H-84	2.0	8	444	437	23,400
		M-84	1.7	8	371	365	23,300
		L-42	1.2	14	234	231	41,500
	Subtotal			30	1,049	1,033	88,200
	PFB	H-84	2.6	32	2,053	2,014	78,000
		M-56	1.9	24	750	736	59,700
		L-42	1.5	14	270	266	34,900
	Subtotal			70	3,073	3,016	172,600
	LWB–21 total			**271**	**8,605**	**8,459**	**380,700**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U17. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-17[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–26	Seed	L-70	4.2	16	390	383	11,300
		L-42	4.2	32	467	460	23,000
	Subtotal			48	857	843	34,300
	Standard blanket	H-84	2.0	8	444	437	23,400
		M-84	1.7	16	742	729	46,600
		M-56	1.7	16	496	488	47,400
		L-42	1.2	5	84	82	14,700
	Subtotal			45	1,766	1,736	132,100
	PFB[e]	H-84	2.6	24	1,541	1,511	58,400
		H-70	2.6	13	697	682	31,700
		M-56	1.9	32	1,000	983	79,600
		L-42	1.5	18	347	342	44,800
	Subtotal			87	3,585	3,518	214,500
	LWB–26 total			**180**	**6,208**	**6,097**	**380,900**
LWB–23	Seed	H-84	5.2	48	1,686	1,659	33,100
		L-70	4.2	15	365	359	10,500
		L-56	4.2	64	1,247	1,224	45,600
		L-42	4.2	32	467	459	23,000
	Subtotal			159	3,765	3,701	112,200
	Standard blanket	M-84	1.7	8	370	364	23,300
	PFB	H-84	2.6	96	6,173	6,049	234,700
	LWB–23 total			**263**	**10,308**	**10,114**	**370,200**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U18. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-18[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–33	Seed	H-84	5.2	48	1,688	1,660	33,200
	Standard blanket	H-84	2.0	24	1,334	1,311	70,200
		H-70	2.0	24	1,110	1,089	69,900
		M-56	1.7	15	464	457	44,300
		L-42	1.2	19	318	313	56,100
	Subtotal			82	3,226	3,170	240,500
	PFB[e]	H-70	2.6	16	857	840	39,000
		M-56	1.9	16	500	491	39,800
		L-42	1.5	16	308	303	39,800
	Subtotal			48	1,665	1,634	118,600
	LWB–33 total			**178**	**6,579**	**6,464**	**392,300**
LWB–29	Seed	L-70	4.2	16	389	383	11,300
		L-42	4.2	47	683	671	33,800
	Subtotal			63	1,072	1,054	45,100
	Standard blanket	H-84	2.0	24	1,335	1,312	70,300
		M-84	1.7	26	1,207	1,186	75,700
		M-56	1.7	16	495	487	47,200
		L-42	1.2	40	670	655	117,800
	Subtotal			106	3,707	3,640	311,000
	PFB	H-70	2.6	8	428	420	19,500
		M-56	1.9	8	250	246	19,900
	Subtotal			16	678	666	39,400
	LWB–29 total			**185**	**5,457**	**5,360**	**395,500**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U19. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-19[a]

Stored shipping canister[b]	Module type	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–22	Seed	H-84	5.2	79	2,773	2,728	54,600
		H-70	4.2	13	317	311	9,100
		L-56	4.2	54	1,049	1,032	38,400
		L-42	4.2	152	2,218	2,181	109,400
	Subtotal			298	6,357	6,252	211,500
	Standard blanket	H-70	2.0	6	277	272	17,000
		M-84	1.7	17	788	776	50,000
		L-42	1.2	8	134	132	23,700
	Subtotal			31	1,199	1,180	90,700
	PFB[e]	H-84	2.6	7	450	441	17,200
		H-70	2.6	8	429	420	19,500
	Subtotal			15	879	861	36,700
	Reflector	Thoria (ThO_2)		2	0	0	12,100
	LWB–22 total			**346**	**8,435**	**8,293**	**351,000**
LWB–20	Standard blanket	H-84	2.0	16	888	872	47,000
		M-84	1.7	8	371	365	23,000
	Subtotal			24	1,259	1,237	70,000
	PFB	H-84	2.6	56	3,592	3,527	137,000
		H-70	2.6	39	2,097	2,048	95,000
		L-42	1.5	31	597	589	77,000
	Subtotal			126	6,286	6,164	309,000
	LWB–20 total			**150**	**7,545**	**7,401**	**379,000**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]For production rods, the rod stratum are generally grouped as "High" (H), "Medium" (M), and "Low" (L) based on their intended positions in the LWBR core. The number following this designation is the rod length in inches.

[d]Measured by [^{233}U (g) / ($UO_2 + ThO_2$) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

Table B.5.U20. Characteristics of unirradiated LWBR fuel in INEEL—ICPP dry storage vault U-20[a]

Stored shipping canister[b]	Module type (Fuel type)	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Production fuel				
LWB–42	Seed	H-84	5.2	5	176	173	3,400
		L-70	4.2	3	73	72	2,100
		L-56	4.2	3	58	57	2,100
		L-42	4.2	10	146	143	7,200
	Subtotal			21	453	445	14,800
	Standard blanket	H-84	2.0	3	166	163	8,700
		H-70	2.0	4	185	183	11,600
		M-84	1.7	3	139	137	8,700
		M-56	1.7	4	123	121	11,800
		L-42	1.2	5	83	82	14,700
	Subtotal			19	696	686	55,500
	PFB[e]	H-84	2.6	5	321	315	12,100
		H-70	2.6	4	214	210	9,600
		M-84	1.9	3	142	139	7,300
		M-56	1.9	4	125	123	10,000
		L-42	1.5	5	96	95	12,300
	Subtotal			21	898	882	51,300
	Reflector	Thoria (ThO_2)		4	0	0	24,300
	Total			**65**	**2,047**	**2,013**	**145,900**
			Proof of breeding (POB) fuel				
	Seed	POB rods		8	120	117	5,700
		Reflector rods		2			1,500
	Subtotal			10	120	117	7,200
	Standard blanket	POB rods		4	95	93	11,600
		Reflector rods		2			5,900
	Subtotal			6	95	93	17,500
	PFB	POB rods		6	159	155	14,800
		Reflector rods		2			5,000
	Subtotal			8	159	155	19,800
	Reflector	Thoria		4	202	197	24,000
	Total			**28**	**576**	**562**	**68,500**

Table B.5.U20 (continued)

Stored shipping canister[b]	Module type (Fuel type)	Rod stratum[c]	^{233}U, wt %[d]	Number of rods	Mass, g		
					Total U	^{233}U	Th
			Miscellaneous				
LWB–42 (contd.)	Misc. rods	GRIP II		8	74	73	3,200
		Defective		2	2	2	*f*
		Pifag		1	4	4	200
		SS rods (BP)[g]		12	516	506	20,900
		SS rods (SP)[h]		6	182	179	3,000
	Subtotal			29	778	764	27,300
	LWB-42 total			**122**	**3,401**	**3,339**	**241,700**
LWB-41	Misc. rods	SS rods (SP)[i]		53	2,116	2,083	36,000
		SS rods (PFB)[j]		24	1,540	1,519	53,100
		SS rods (BP)[k]		60	3,325	3,267	164,600
		SS rods (RP)[l]		1	41	40	4,100
	LWB-41 total			**138**	**7,030**	**6,909**	**262,800**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1101 [Mitchell, Semans, and Smith (October 1974)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]There are two stacked canisters in this dry storage vault. The canister stacked on top is listed first.

[c]Rod stratum not available for detailed cell rods. The "Type" listed is based on, and associated with, the reported ^{233}U wt %.

[d]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[e]PFB = power-flattening blanket.

[f]Negligible amount.

[g]Mixture of short rods and stainless steel (SS)-cladded rods with blanket pellets (BP).

[h]Mixture of short rods and SS-cladded rods with seed pellets (SP).

[i]SS-cladded rods with seed pellets.

[j]SS-cladded rods with PFB pellets.

[k]SS-cladded rods with blanket pellets (BP).

[l]SS-cladded rods with reflector pellets (RP).

Table B.5.U22. Characteristics of spare unirradiated LWBR seed module stored in INEEL—ICPP dry storage vault U-22[a]

Module type	Rod stratum[b]	^{233}U, wt %[c]	Number of rods	Mass, g		
				Total U	^{233}U	Th
Seed III-5[d]	H-84	5.2	331	11,642	11,452	181,923
	L-70	4.2	66	1,606	1,580	31,457
	L-56	4.2	72	1,402	1,379	27,455
	L-42	4.2	150	2,188	2,152	42,844
	Thoria (ThO_2)	0.0	0	0	0	150,321
	Total		**619**	**16,838**	**16,563**	**434,000**

[a]Based on Detrick Apr. 8, 1998, and refs. WAPD-EA-318 [Detrick (May 6, 1997)]; WAPD-TM-1600 [Atherton, R. et al. (October 1987)]; and WAPD-TM-1612 [Schick et al. (September 1987)].

[b]Identified by ^{233}U concentration (H = high, M = medium, and L = low) and binary fuel (UO_2 and ThO_2) stack length in inches.

[c]Measured by [^{233}U (g) / (UO_2 + ThO_2) (g)] × 100%. All binary fuel pellets in any given rod have the same concentration of fissile uranium.

[d]Stored in a liner in Dry Storage Vault U-22, which is at the southern end of the INEEL/ICPP unirradiated LWBR Fuel Storage Area.

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